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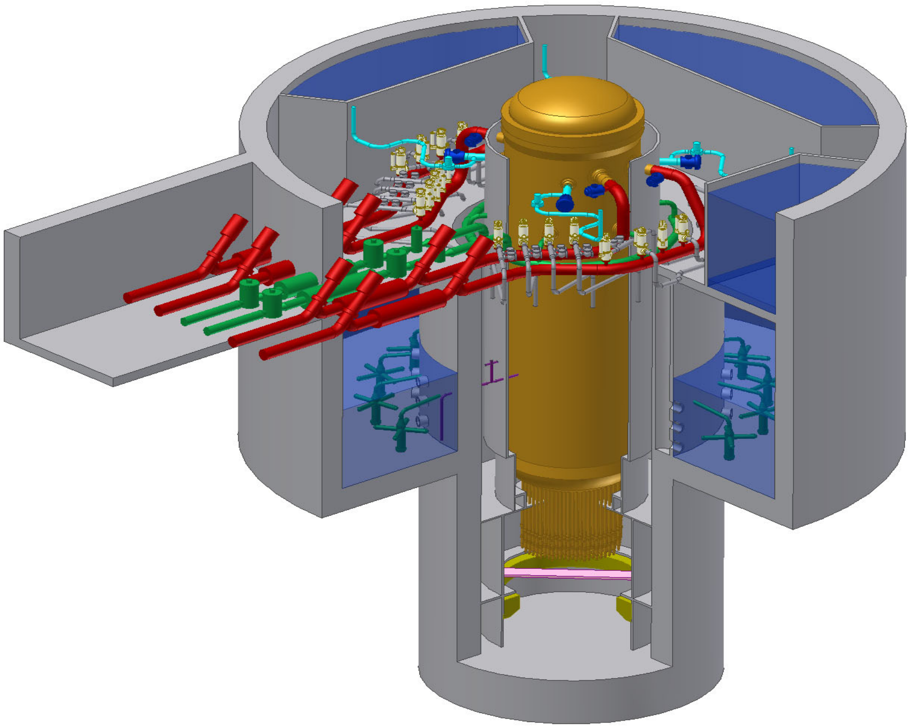

ESBWR Design Control Document

Tier 2

Chapter 15

Safety Analyses

(Conditional Release - pending closure of design verifications)

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Global Abbreviations And Acronyms List

Term	Definition
10 CFR	Title 10, Code of Federal Regulations
A/D	Analog-to-Digital
AASHTO	American Association of Highway and Transportation Officials
AB	Auxiliary Boiler
ABS	Auxiliary Boiler System
ABWR	Advanced Boiling Water Reactor
ac / AC	Alternating Current
AC	Air Conditioning
ACF	Automatic Control Function
ACI	American Concrete Institute
ACS	Atmospheric Control System
AD	Administration Building
ADS	Automatic Depressurization System
AEC	Atomic Energy Commission
AFIP	Automated Fixed In-Core Probe
AGMA	American Gear Manufacturer's Association
AHS	Auxiliary Heat Sink
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
AL	Analytical Limit
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AOV	Air Operated Valve
API	American Petroleum Institute
APRM	Average Power Range Monitor
APR	Automatic Power Regulator
APRS	Automatic Power Regulator System
ARI	Alternate Rod Insertion
ARMS	Area Radiation Monitoring System
ASA	American Standards Association
ASD	Adjustable Speed Drive
ASHRAE	American Society of Heating, Refrigerating, and Air Conditioning Engineers
ASME	American Society of Mechanical Engineers
AST	Alternate Source Term
ASTM	American Society of Testing Methods
AT	Unit Auxiliary Transformer

Term	Definition
ATLM	Automated Thermal Limit Monitor
ATWS	Anticipated Transients Without Scram
AV	Allowable Value
AWS	American Welding Society
AWWA	American Water Works Association
B&PV	Boiler and Pressure Vessel
BAF	Bottom of Active Fuel
BHP	Brake Horse Power
BOP	Balance of Plant
BPU	Bypass Unit
BPV	Bypass Valve
BPWS	Banked Position Withdrawal Sequence
BRE	Battery Room Exhaust
BRL	Background Radiation Level
BTP	NRC Branch Technical Position
BTU	British Thermal Unit
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CAV	Cumulative absolute velocity
C&FS	Condensate and Feedwater System
C&I	Control and Instrumentation
C/C	Cooling and Cleanup
CB	Control Building
CBGAHVS	Control Building General Area
CBHVAC	Control Building HVAC
CBHVS	Control Building Heating, Ventilation and Air Conditioning System
CCI	Core-Concrete Interaction
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CIRC	Circulating Water System
CIS	Containment Inerting System
CIV	Combined Intermediate Valve
CLAVS	Clean Area Ventilation Subsystem of Reactor Building HVAC
CM	Cold Machine Shop
CMS	Containment Monitoring System
CMU	Control Room Multiplexing Unit
COL	Combined Operating License
COLR	Core Operating Limits Report
CONAVS	Controlled Area Ventilation Subsystem of Reactor Building HVAC
CPR	Critical Power Ratio

Term	Definition
CPS	Condensate Purification System
CPU	Central Processing Unit
CR	Control Rod
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CRDH	Control Rod Drive Housing
CRDHS	Control Rod Drive Hydraulic System
CRGT	Control Rod Guide Tube
CRHA	Control Room Habitability Area
CRHAHVS	Control Room Habitability Area HVAC Sub-system
CRT	Cathode Ray Tube
CS&TS	Condensate Storage and Transfer System
CSDM	Cold Shutdown Margin
CS / CST	Condensate Storage Tank
CT	Main Cooling Tower
CTVCF	Constant Voltage Constant Frequency
CUF	Cumulative usage factor
CWS	Chilled Water System
D-RAP	Design Reliability Assurance Program
DAC	Design Acceptance Criteria
DAW	Dry Active Waste
DBA	Design Basis Accident
DBE	Design Basis Event
dc / DC	Direct Current
DCS	Drywell Cooling System
DCIS	Distributed Control and Information System
DEPSS	Drywell Equipment and Pipe Support Structure
DF	Decontamination Factor
D/F	Diaphragm Floor
DG	Diesel-Generator
DHR	Decay Heat Removal
DM&C	Digital Measurement and Control
DOF	Degree of freedom
DOI	Dedicated Operators Interface
DOT	Department of Transportation
dPT	Differential Pressure Transmitter
DPS	Diverse Protection System
DPV	Depressurization Valve
DR&T	Design Review and Testing
DS	Independent Spent Fuel Storage Installation

Term	Definition
DTM	Digital Trip Module
DW	Drywell
EB	Electrical Building
EBAS	Emergency Breathing Air System
EBHV	Electrical Building HVAC
ECCS	Emergency Core Cooling System
E-DCIS	Essential DCIS (Distributed Control and Information System)
EDO	Environmental Qualification Document
EFDS	Equipment and Floor Drainage System
EFPY	Effective full power years
EFU	Emergency Filter Unit
EHC	Electrohydraulic Control (Pressure Regulator)
ENS	Emergency Notification System
EOC	Emergency Operations Center
EOC	End of Cycle
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedures
EPDS	Electric Power Distribution System
EPG	Emergency Procedure Guidelines
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
ERICP	Emergency Rod Insertion Control Panel
ERIP	Emergency Rod Insertion Panel
ESF	Engineered Safety Feature
ETS	Emergency Trip System
FAC	Flow-Accelerated Corrosion
FAPCS	Fuel and Auxiliary Pools Cooling System
FATT	Fracture Appearance Transition Temperature
FB	Fuel Building
FBHV	Fuel Building HVAC
FCI	Fuel-Coolant Interaction
FCM	File Control Module
FCS	Flammability Control System
FCU	Fan Cooling Unit
FDDI	Fiber Distributed Data Interface
FFT	Fast Fourier Transform
FFWTR	Final Feedwater Temperature Reduction
FHA	Fire Hazards Analysis
FIV	Flow-Induced Vibration
FMCRD	Fine Motion Control Rod Drive

Term	Definition
FMEA	Failure Modes and Effects Analysis
FPS	Fire Protection System
FO	Diesel Fuel Oil Storage Tank
FOAKE	First-of-a-Kind Engineering
FPE	Fire Pump Enclosure
FTDC	Fault-Tolerant Digital Controller
FTS	Fuel Transfer System
FW	Feedwater
FWCS	Feedwater Control System
FWS	Fire Water Storage Tank
GCS	Generator Cooling System
GDC	General Design Criteria
GDCS	Gravity-Driven Cooling System
GE	General Electric Company
GE-NE	GE Nuclear Energy
GEN	Main Generator System
GETAB	General Electric Thermal Analysis Basis
GL	Generic Letter
GM	Geiger-Mueller Counter
GM-B	Beta-Sensitive GM Detector
GSIC	Gamma-Sensitive Ion Chamber
GSOS	Generator Sealing Oil System
GWSR	Ganged Withdrawal Sequence Restriction
HAZ	Heat-Affected Zone
HCU	Hydraulic Control Unit
HCW	High Conductivity Waste
HDVS	Heater Drain and Vent System
HEI	Heat Exchange Institute
HELB	High Energy Line Break
HEP	Human error probability
HEPA	High Efficiency Particulate Air/Absolute
HFE	Human Factors Engineering
HFF	Hollow Fiber Filter
HGCS	Hydrogen Gas Cooling System
HIC	High Integrity Container
HID	High Intensity Discharge
HIS	Hydraulic Institute Standards
HM	Hot Machine Shop & Storage
HP	High Pressure
HPNSS	High Pressure Nitrogen Supply System

Term	Definition
HPT	High-pressure turbine
HRA	Human Reliability Assessment
HSI	Human-System Interface
HSSS	Hardware/Software System Specification
HVAC	Heating, Ventilation and Air Conditioning
HVS	High Velocity Separator
HWC	Hydrogen Water Chemistry
HWCS	Hydrogen Water Chemistry System
HWS	Hot Water System
HX	Heat Exchanger
I&C	Instrumentation and Control
I/O	Input/Output
IAS	Instrument Air System
IASCC	Irradiation Assisted Stress Corrosion Cracking
IBC	International Building Code
IC	Ion Chamber
IC	Isolation Condenser
ICD	Interface Control Diagram
ICS	Isolation Condenser System
IE	Inspection and Enforcement
IEB	Inspection and Enforcement Bulletin
IED	Instrument and Electrical Diagram
IEEE	Institute of Electrical and Electronic Engineers
IFTS	Inclined Fuel Transfer System
IGSCC	Intergranular Stress Corrosion Cracking
IIS	Iron Injection System
ILRT	Integrated Leak Rate Test
IOP	Integrated Operating Procedure
IMC	Induction Motor Controller
IMCC	Induction Motor Controller Cabinet
IRM	Intermediate Range Monitor
ISA	Instrument Society of America
ISI	In-Service Inspection
ISLT	In-Service Leak Test
ISM	Independent Support Motion
ISMA	Independent Support Motion Response Spectrum Analysis
ISO	International Standards Organization
ITA	Inspections, Tests or Analyses
ITAAC	Inspections, Tests, Analyses and Acceptance Criteria
ITA	Initial Test Program

Term	Definition
LAPP	Loss of Alternate Preferred Power
LCO	Limiting Conditions for Operation
LCW	Low Conductivity Waste
LD	Logic Diagram
LDA	Lay down Area
LD&IS	Leak Detection and Isolation System
LERF	Large early release frequency
LFCV	Low Flow Control Valve
LHGR	Linear Heat Generation Rate
LLRT	Local Leak Rate Test
LMU	Local Multiplexer Unit
LO	Dirty/Clean Lube Oil Storage Tank
LOCA	Loss-of-Coolant-Accident
LOFW	Loss-of-feedwater
LOOP	Loss of Offsite Power
LOPP	Loss of Preferred Power
LP	Low Pressure
LPCI	Low Pressure Coolant Injection
LPCRD	Locking Piston Control Rod Drive
LPMS	Loose Parts Monitoring System
LPRM	Local Power Range Monitor
LPSP	Low Power Setpoint
LWMS	Liquid Waste Management System
MAAP	Modular Accident Analysis Program
MAPLHGR	Maximum Average Planar Linear Head Generation Rate
MAPRAT	Maximum Average Planar Ratio
MBB	Motor Built-In Brake
MCC	Motor Control Center
MCES	Main Condenser Evacuation System
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MCRP	Main Control Room Panel
MELB	Moderate Energy Line Break
MLHGR	Maximum Linear Heat Generation Rate
MMI	Man-Machine Interface
MMIS	Man-Machine Interface Systems
MOV	Motor-Operated Valve
MPC	Maximum Permissible Concentration
MPL	Master Parts List
MS	Main Steam

Term	Definition
MSIV	Main Steam Isolation Valve
MSL	Main Steamline
MSLB	Main Steamline Break
MSLBA	Main Steamline Break Accident
MSR	Moisture Separator Reheater
MSV	Mean Square Voltage
MT	Main Transformer
MTTR	Mean Time To Repair
MWS	Makeup Water System
NBR	Nuclear Boiler Rated
NBS	Nuclear Boiler System
NCIG	Nuclear Construction Issues Group
NDE	Nondestructive Examination
NE-DCIS	Non-Essential Distributed Control and Information System
NDRC	National Defense Research Committee
NDT	Nil Ductility Temperature
NFPA	National Fire Protection Association
NIST	National Institute of Standard Technology
NICWS	Nuclear Island Chilled Water Subsystem
NMS	Neutron Monitoring System
NOV	Nitrogen Operated Valve
NPHS	Normal Power Heat Sink
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NRHX	Non-Regenerative Heat Exchanger
NS	Non-seismic (non-seismic Category I)
NSSS	Nuclear Steam Supply System
NT	Nitrogen Storage Tank
NTSP	Nominal Trip Setpoint
O&M	Operation and Maintenance
O-RAP	Operational Reliability Assurance Program
OBCV	Overboard Control Valve
OBE	Operating Basis Earthquake
OGS	Offgas System
OHLHS	Overhead Heavy Load Handling System
OIS	Oxygen Injection System
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLU	Output Logic Unit
OOS	Out-of-service
ORNL	Oak Ridge National Laboratory

Term	Definition
OSC	Operational Support Center
OSHA	Occupational Safety and Health Administration
OSI	Open Systems Interconnect
P&ID	Piping and Instrumentation Diagram
PA/PL	Page/Party-Line
PABX	Private Automatic Branch (Telephone) Exchange
PAM	Post Accident Monitoring
PAR	Passive Autocatalytic Recombiner
PAS	Plant Automation System
PASS	Post Accident Sampling Subsystem of Containment Monitoring System
PCC	Passive Containment Cooling
PCCS	Passive Containment Cooling System
PCT	Peak cladding temperature
PCV	Primary Containment Vessel
PFD	Process Flow Diagram
PGA	Peak Ground Acceleration
PGCS	Power Generation and Control Subsystem of Plant Automation System
PH	Pump House
PL	Parking Lot
PM	Preventive Maintenance
PMCS	Performance Monitoring and Control Subsystem of NE-DCIS
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PQCL	Product Quality Check List
PRA	Probabilistic Risk Assessment
PRMS	Process Radiation Monitoring System
PRNM	Power Range Neutron Monitoring
PS	Plant Stack
PSD	Power Spectra Density
PSS	Process Sampling System
PSWS	Plant Service Water System
PT	Pressure Transmitter
PWR	Pressurized Water Reactor
QA	Quality Assurance
RACS	Rod Action Control Subsystem
RAM	Reliability, Availability and Maintainability
RAPI	Rod Action and Position Information
RAT	Reserve Auxiliary Transformer
RB	Reactor Building
RBC	Rod Brake Controller

Term	Definition
RBCC	Rod Brake Controller Cabinet
RBCWS	Reactor Building Chilled Water Subsystem
RBHV	Reactor Building HVAC
RBS	Rod Block Setpoint
RBV	Reactor Building Vibration
RC&IS	Rod Control and Information System
RCC	Remote Communication Cabinet
RCCV	Reinforced Concrete Containment Vessel
RCCWS	Reactor Component Cooling Water System
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RDA	Rod Drop Accident
RDC	Resolver-to-Digital Converter
REPAVS	Refueling and Pool Area Ventilation Subsystem of Fuel Building HVAC
RFP	Reactor Feed Pump
RG	Regulatory Guide
RHR	Residual heat removal (function)
RHX	Regenerative Heat Exchanger
RMS	Root Mean Square –or- Radiation Monitoring Subsystem
RMU	Remote Multiplexer Unit
RO	Reverse Osmosis
ROM	Read-only Memory
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RRPS	Reference Rod Pull Sequence
RSM	Rod Server Module
RSPC	Rod Server Processing Channel
RSS	Remote Shutdown System
RSSM	Reed Switch Sensor Module
RSW	Reactor Shield Wall
RTIF	Reactor Trip and Isolation Function(s)
RT_{NDT}	Reference Temperature of Nil-Ductility Transition
RTP	Reactor Thermal Power
RW	Radwaste Building
RWBCR	Radwaste Building Control Room
RWBGA	Radwaste Building General Area
RWBHVAC	Radwaste Building HVAC
RWCU/SDC	Reactor Water Cleanup/Shutdown Cooling
RWE	Rod Withdrawal Error
RWM	Rod Worth Minimizer

Term	Definition
SA	Severe Accident
SAR	Safety Analysis Report
SB	Service Building
S/C	Digital Gamma-Sensitive GM Detector
SC	Suppression Chamber
S/D	Scintillation Detector
S/DRSRO	Single/Dual Rod Sequence Restriction Override
S/N	Signal-to-Noise
S/P	Suppression Pool
SAS	Service Air System
SB&PC	Steam Bypass and Pressure Control System
SBO	Station Blackout
SBWR	Simplified Boiling Water Reactor
SCEW	System Component Evaluation Work
SCRRI	Selected Control Rod Run-in
SDC	Shutdown Cooling
SDM	Shutdown Margin
SDS	System Design Specification
SEOA	Sealed Emergency Operating Area
SER	Safety Evaluation Report
SF	Service Water Building
SFP	Spent fuel pool
SIL	Service Information Letter
SIT	Structural Integrity Test
SIU	Signal Interface Unit
SJAE	Steam Jet Air Ejector
SLC	Standby Liquid Control
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SMU	SSLC Multiplexing Unit
SOV	Solenoid Operated Valve
SP	Setpoint
SPC	Suppression Pool Cooling
SPDS	Safety Parameter Display System
SPTMS	Suppression Pool Temperature Monitoring Subsystem of Containment Monitoring System
SR	Surveillance Requirement
SRM	Source Range Monitor
SRNM	Startup Range Neutron Monitor
SRO	Senior Reactor Operator
SRP	Standard Review Plan

Term	Definition
SRS	Software Requirements Specification
SRSRO	Single Rod Sequence Restriction Override
SRSS	Sum of the squares
SRV	Safety Relief Valve
SRVDL	Safety relief valve discharge line
SSAR	Standard Safety Analysis Report
SSC(s)	Structure, System and Component(s)
SSE	Safe Shutdown Earthquake
SSLC	Safety System Logic and Control
SSPC	Steel Structures Painting Council
ST	Spare Transformer
STI	Startup Test Instruction
STP	Sewage Treatment Plant
STRAP	Scram Time Recording and Analysis Panel
STRP	Scram Time Recording Panel
SV	Safety Valve
SWH	Static water head
SWMS	Solid Waste Management System
SY	Switch Yard
TAF	Top of Active Fuel
TASS	Turbine Auxiliary Steam System
TB	Turbine Building
TBCE	Turbine Building Compartment Exhaust
TEAS	Turbine Building Air Supply
TBE	Turbine Building Exhaust
TBLOE	Turbine Building Lube Oil Area Exhaust
TBS	Turbine Bypass System
TBHV	Turbine Building HVAC
TBV	Turbine Bypass Valve
TC	Training Center
TCCWS	Turbine Component Cooling Water System
TCS	Turbine Control System
TCV	Turbine Control Valve
TDH	Total Developed Head
TEMA	Tubular Exchanger Manufacturers' Association
TFSP	Turbine first stage pressure
TG	Turbine Generator
TGSS	Turbine Gland Seal System
THA	Time-history accelerograph
TLOS	Turbine Lubricating Oil System

Term	Definition
TLU	Trip Logic Unit
TMI	Three Mile Island
TMSS	Turbine Main Steam System
TRM	Technical Requirements Manual
TS	Technical Specification(s)
TSC	Technical Support Center
TSI	Turbine Supervisory Instrument
TSV	Turbine Stop Valve
TTWFATBV	Turbine trip with failure of all bypass valves
UBC	Uniform Building Code
UHS	Ultimate heat sink
UL	Underwriter's Laboratories Inc.
UPS	Uninterruptible Power Supply
USE	Upper Shelf Energy
USM	Uniform Support Motion
USMA	Uniform support motion response spectrum analysis
USNRC	United States Nuclear Regulatory Commission
USS	United States Standard
UV	Ultraviolet
V&V	Verification and Validation
Vac / VAC	Volts Alternating Current
Vdc / VDC	Volts Direct Current
VDU	Video Display Unit
VW	Vent Wall
VWO	Valves Wide Open
WD	Wash Down Bays
WH	Warehouse
WS	Water Storage
WT	Water Treatment
WW	Wetwell
XMFR	Transformer
ZPA	Zero period acceleration

15. SAFETY ANALYSES

This chapter addresses all ESBWR plant safety analyses. The details of most of the safety analyses are contained within this chapter, however, per the Regulatory Guide 1.70 format, some safety analyses are addressed in detail in other DCD Tier 2 chapters (e.g. emergency core cooling system [ECCS] performance is addressed within Section 6.3). For those safety analyses not addressed in detail in Chapter 15, references are provided to their locations within Tier 2.

15.0 ANALYTICAL APPROACH

In this chapter, the effects of anticipated process disturbances and postulated component failures are examined to determine their consequences and to evaluate plant capabilities to control or accommodate such failures and events. System response analyses are based upon the core loading shown in Figure 4.3-1, and is used to identify the limiting events for the ESBWR. Other fuel designs and core loading patterns, including loading patterns similar to Figure 4.3-2, do not affect the sensitivities demonstrated by this study. Evaluation of these limiting events for each plant fuel cycle ensures that the criteria in Appendix 4B are met.

GE has developed a unique systematic approach to plant safety consistent with the GE boiling water reactor (BWR) technology base. The key to the GE approach to plant safety is the Section 15.1 Nuclear Safety Operational Analysis (NSOA). A NSOA is a system level qualitative failure modes and effects analysis (FMEA) of plant protective functions to show compliance with the events addressed in Chapter 15. Key inputs into the NSOA are derived from the applicable regulations, through industry codes and standards, and the plant safety analyses.

In Section 15.1, all unacceptable safety results and all required safety actions are identified. In addition, an evaluation of the entire spectrum of events is consistently carried out for the ESBWR to demonstrate that a consistent level of safety has been attained.

The NSOA acceptance criteria are based on the Title 10 of the Code of Federal Regulations (10 CFR regulations) and the NUREG-0800 Standard Review Plan (SRP) acceptance criteria.

The starting point for the NSOA is the establishment of unacceptable safety results. This concept enables the results of any safety analysis to be compared to applicable criteria. Unacceptable safety results represent an extension of the nuclear design criteria for plant systems and components that are used as the basis for system design. The unacceptable safety results have been selected so that they are consistent with applicable regulations and industry codes and standards.

The focal point of the NSOA is the event analysis, in which all essential protection sequences are evaluated until all required safety actions are successfully completed. The event analysis identifies all required front-line safety systems and their essential auxiliaries.

The full spectrum of initial conditions limited by the constraints placed on planned operation is evaluated. All events are analyzed until a stable condition is obtained. This ensures that the event being evaluated does not have an unevaluated long-term consideration.

In the event analysis, all essential systems, operator actions and limits to satisfy the required safety actions are identified. Limits are derived only for those parameters continuously available to the operator. Credit for operator action is taken only when an operator can be reasonably expected to perform the required action based on the information available to him.

In the NSOA, a complete and consistent set of safety actions (i.e., those required to prevent unacceptable results) has been developed. For all of the events that are evaluated, a single-failure-proof path to plant shutdown is identified. The application of the 10 CFR 50, Appendix A single-failure criterion (SFC) to these events is imposed as an additional measure of conservatism in the NSOA process.

15.0.1 Classification and Selection of Events

From Reference 1, the classification of events for the ESBWR is primarily based on the classifications and terms used in the 10 CFR regulations because:

- the 10 CFR regulations have authority over all other document types;
- the non-accident abnormal event classifications within the SRP are inconsistently used;
- the non-accident abnormal event classifications within Regulatory Guide (RG) 1.70 are inconsistently used;
- the classifications of non-accident abnormal event classifications between the SRP and RG 1.70 are inconsistent;
- both sets of non-accident abnormal event classifications in the SRP and RG 1.70 are not consistent with the abnormal event classifications in the 10 CFR regulations;
- all versions of abnormal event categories are not clearly defined in the SRP and RG 1.70;
- the 10 CFR regulations do specifically define an Anticipated Operational Occurrence (AOO), Loss-of-Coolant Accident (LOCA), Anticipated Transient Without Scram (ATWS), normal operation, design basis events, and a number of associated terms; and
- the use of terms is more consistent within the 10 CFR regulations than in the SRP or RG 1.70.

The most recently certified BWR (i.e., the ABWR) licensing documents are used for additional guidance.

The design basis events (DBEs) in the 10 CFR regulations assume an initiating event (and any resultant failures) with or without a single active component failure or operator error. The postulating of design basis events that assume a failure beyond the SFC or a common-mode failure is not specifically required by the 10 CFR regulations. However, the 10 CFR regulations do require evaluations of three specific event scenarios, i.e., Safe Shutdown Fire, Station Blackout (SBO) and ATWS, and some of these event scenarios do assume failures beyond the single failure criterion (SFC) and/or common-mode failures. Therefore, these events should not be classified as DBEs. However, their safety analyses are included in the ESBWR DCD.

Based on Table 3-1 of ANSI/ANS-52.1, DBEs should have annual probabilities $\geq 10^{-6}$. Therefore, any event with an annual probability of $< 10^{-6}$ is not considered credible and is not classified as a DBE.

The 10CFR regulations, SRP and RG 1.70 postulate events that (for the ESBWR with its advanced design features and additional redundancy) require failures beyond the SFC and/or require common-mode failures. Those events are included in the ESBWR DCD, but not as DBEs.

Per the 10 CFR regulations, AOOs are expected to occur once in a plant's lifetime, while accidents are low probability events that are not expected to occur during a plant's lifetime. Because the ESBWR has a design life is 60 years, any abnormal event that has an annual probability of occurrence ≥ 1/60 could be classified as an AOO. However, historically, a value of > 1/100 has been used.

Based on the 10 CFR regulations, the SRP or an NRC reviewed Licensing Topical Report (LTR), the safety analysis acceptance criteria for each of the special events is developed on an event-specific basis.

The 10 CFR regulations consistently refer to any failure of a fission product barrier that results in an offsite radiological consequence as an accident.

15.0.1.1 Approach For Determining Event Classifications

(1) Per the 10 CFR regulations, the 10 CFR 50 App. A definitions, GDC, the 10 CFR 50.49 design basis event definition, SRP 6.1.1, SRP 15.0.1, RG 1.183 and guidance from events addressed in the SRP;

   a. divide the types of events as DBEs, and by exclusion, all other events as special events;

   b. provide the basis for which events should be classified as AOOs;

   c. provide the basis for a (non-AOO and non-accident) event classification for events with lower probabilities than AOOs, but for conservatism have historically not been treated or classified as accidents; and

   d. generate the criterion for determining which type of accidents shall be classified as design basis accidents (DBAs), and by exclusion, all other accidents are not DBAs.

(2) Per the regulatory definition of an AOO (event probability), historical information and guidance from the SRP determine specific criteria for classifying events as AOOs.

(3) Based on (a) the 10 CFR regulations associating accidents with radiological consequences, (b) application of SFC, (c) SRP and RG 1.70 guidance for the types of events that should be addressed in Chapter 15, (d) SRP acceptance criteria for transient/AOO events that result in fuel failure, and (e) historical consistently used terms, generate a classification term and criteria for determining non-AOO and non-accidents events, which (a) should be treated as design basis events and (b) result from an initiating event with or without assuming a single active component failure or single operator error. Include this new DBE term in the DBE classifications.

(4) Based on the 10 CFR regulations, SRPs, RG 1.183 and the ABWR DCD portions associated accidents, generate a definition for an accident.

(5) Based on (a) reviewing the 10 CFR regulations that have added other abnormal events (e.g., ATWS, SBO, Safe Shutdown Fire), (b) that DBEs do not include common-mode failures and/or additional failure(s) beyond the SFC, (c) reviewing the SRP events that include common-mode failures and/or failure(s) beyond the SFC, and (d) historically evaluated non-DBE events and used associated classification terms, generate classification term for non-DBEs that are addressed in the DCD Chapter 15.

15.0.1.2 Results of Event Classification Determinations

Table 15.0-1 provides the results of the event classifications in the form of a determination criterion vs. event classification matrix. Table 15.0-1 is based on the results from the following evaluation.

(1) a. Per 10 CFR 50.49, and the fact that the SRP treats all postulated abnormal initiating events with or without assuming a single active component failure or single operator error as if they are all design basis events, the following are classified as design basis events:

- normal operation, including AOOs;
- accidents [see (3) for additional details];
- design basis accidents;
- external events; and
- natural phenomena,

(1) b. AOOs, by definition, are classified as part of normal operations, do not have radiological consequences (except if in combination with an additional single active component failure or single operator error), have more restrictive acceptance criteria (e.g., GDC 10 or 10CFR 20 vs. 10 CFR 50.34) than accidents, and thus, are not accidents and shall not be treated as accidents.

(1) c. The classification term for events with lower probabilities than AOOs, but for conservatism should be not treated as accidents should be based on the ABWR.

(1) d. Except for AOOs, the 10 CFR regulations, SRP and RG 1.70 do not explicitly or implicitly apply any quantitative event frequency criterion for defining any other abnormal event classification. Therefore, event frequencies should not be used to determine accident type event classifications.

SRP 6.1.1, SRP 15.0.1 and RG 1.183 are consistent in categorization of DBAs. A DBA *is an accident postulated and analyzed to confirm the adequacy of a plant engineered safety feature*.

By exclusion, all other accidents are not classified as DBAs.

(2) An AOO *is any abnormal event that has an event probability of ≥ 1/100 per year*.

(3) Based on the ABWR DCD Tier 2 Subsection 15A.2.2, the other (non-AOO and non-DBA) postulated abnormal events are classified as "*infrequent events*." An infrequent event *is defined as a DBE (with or without assuming a single active component failure or single operator error) with probability of occurrence of < 1/100 per year, and a radiological consequence less than an accident.*

(4) The other (non-AOO and non-infrequent incident) DBEs should be classified as accidents with DBAs as a subset. An accident *is defined as a postulated DBE that is not expected to occur during the lifetime of a plant, which equates to either an ASME Code Service Level C or D incident, and results in radioactive material releases with calculated doses comparable to (but not to exceed) the 10 CFR 50.34(a) exposures.*

(5) Historically, non-DBEs that are evaluated in BWR safety analysis reports or DCD have been termed as "special events." As no better term has been specified in a regulatory document, it is judged reasonable to maintain that term in the ESBWR DCD.

Special events

a. *Are not included as design basis events in 10 CFR 50.49, and*

i. *are postulated in the 10 CFR regulations to demonstrate some specified prevention, coping or mitigation capabilities, without specifically requiring a radiological evaluation, and/or*

ii. *include a common mode equipment failure or additional failure(s) beyond the SFC.*

b. *Do not include severe accidents or other events that are only evaluated as part of the plant PRA.*

Because of the ESBWR's advanced engineering and additional redundant features, some of the abnormal events for earlier plants are classified differently for the ESBWR.

15.0.2 Abnormal Events To Be Evaluated

In selecting the AOOs to be analyzed as part of the plant safety analysis, the nuclear system parameter variations listed below are considered possible initiating causes of challenges to the fuel or the reactor coolant pressure boundary (RCPB).

- Decrease in Core Coolant Temperature
- Increase in Reactor Pressure
- Increase in Reactor Coolant Inventory
- Decrease in Reactor Coolant Inventory

The AOOs are considered in the ESBWR safety analyses are listed in Table 15.0-2.

The parameter variations listed above include all the effects within the nuclear system (caused by AOOs) that can challenge the integrity of the reactor fuel or RCPB. The variation of any one parameter may cause a change in another parameter. However, for analysis purposes, challenges to barrier integrity are evaluated by groups according to the parameter variation initiating the plant challenge and which typically dominates the event response.

As discussed in Reference 15.0-2 (GESTAR II), and demonstrated in Section 15.2, the following potentially limiting AOOs are re-evaluated for each fuel reload:

- Loss of Feedwater Heating
- Closure of One Turbine Control Valve
- Turbine Trip with Bypass
- Inadvertent Isolation Condenser Initiation
- Loss of Non-Emergency AC to Station Auxiliaries

The infrequent events considered in the ESBWR safety analyses are listed in Table 15.0-2, and are discussed in detail in Section 15.3. The Loss of Feedwater Heating With Failure of Selected Control Rod Run-In is re-evaluated for each fuel reload.

The accidents considered in the ESBWR safety analyses are listed in Table 15.0-2.

The following accidents pose the most limiting challenge to plant design and radiological exposure limits:

- Loss of Coolant Accident (LOCA) Inside Containment
- Main Steamline Break Outside Containment
- Fuel Handling Accident

The LOCA is re-evaluated each reload as part of the process for establishing the core operating limits for new fuel types.

Each of the accidents listed in Table 15.0-2 is discussed in detail in Section 15.4.

The special events evaluated as part of the ESBWR safety analysis are listed in Table 15.0-2, and discussed in detail in Section 15.5. The MSIV Closure With Flux Scram (Overpressure Protection) event will be re-evaluated for each reload, to ensure that the Reactor Coolant System Pressure Safety Limit in the Technical Specifications cannot be exceeded by any Design Basis Event.

15.0.3 Determination of Safety Analysis Acceptance Criteria

Where acceptance criteria are specified in the 10 CFR regulations, those criteria or their equivalent SRP criteria shall be used. However, if an acceptance criterion in the SRP conflicts with the associated acceptance criterion in a regulation, then the criterion specified in the regulation is used. Where an acceptance criterion is not specified in the 10 CFR regulations, then the criterion in the SRP is used. Where an acceptance criterion is not specified in regulations and the SRP, then the criterion in the Final Safety Evaluation Report (FSER) for the ABWR (Reference 15.0-3) or an NRC reviewed LTR shall be used.

A listing of the ESBWR abnormal events and their event classifications and relevant SRP section is provided in Table 15.0-2. Table 15.0-2 is subject to change due to future probabilistic analyses or regulatory considerations, and thus, may be revised in the future.

15.0.3.1 Anticipated Operational Occurrences

To meet the intent of GDC 10, SRP 15.1 and 15.2, detailed acceptance criteria for AOOs both not in combination and in combination with an additional single active component failure (SACF) or single operator error (SOE) are provided. For an AOO, which is not in combination with an additional SACF or SOE, the SRP 15.1 and 15.2 criterion is "Fuel cladding integrity shall be maintained by ensuring that the minimum CPR remains above the MCPR safety limit for BWRs based on acceptable correlations." This is equivalent to the fuel cladding integrity (greater than 99.9% of the fuel rods in the core would be expected to avoid boiling transition) safety limit (SL) included in the ESBWR Technical Specifications (TS).

For an AOO in combination with an additional SACF or SOE, the SRP 15.1 and 15.2 criterion is "fuel failure must be assumed for all rods for which the CPR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding." However, the SRP does not provide a specific radiological acceptance criterion, in the event that fuel cladding failures do occur. As AOOs are part of normal operation, GDC 60 and 10 CFR 20 apply.

The 10 CFR 20.1301(a)(1) 0.1 rem annual dose limit combined with (i.e., subtracting) the 10 CFR 20.1302(b)(2)(ii) 0.05 rem annual limit (for normal airborne releases) is the appropriate radiological acceptance limit for an AOO In Combination With An Additional SACF or SOE (i.e., an AOO with an additional single failure). This position is conservatively based on an assumption that an individual at the exclusion boundary annually recieves 100% of the 10 CFR 20.1302(b)(2)(ii) 0.05 rem annual limit from normal operations (which is conservative, when compared to the 10 CFR 50, Appendix I 10 millirad ALARA annual airborne gamma dose guideline), and applying the 10 CFR 20.1301(a)(1) 0.1 rem annual dose limit. Therefore, the radiological acceptance criterion for an AOO with a single failure should generically be (0.1 - 0.05) 0.05 rem TEDE.

For AOOs, the GDC 15 acceptance criterion is that "The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences." The equivalent criterion in SRP 15.1 and 15.2 is "Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values," which corresponds to the ASME Code Service Level B limit. However, for completeness the Reactor Coolant System Pressure Safety Limit in the Technical Specifications should be addressed.

The SRP provides an AOO related acceptance criterion that is not addressed in GDC 10 or 15, which is "An incident of moderate frequency (i.e., an AOO) should not generate a more serious plant condition without other faults occurring independently."

As shown in Subsection 2.4.2, the ABWR FSER has a nearly equivalent set of AOO acceptance criteria.

Consistent with GDC 38, if an AOO involves Safety/Relief Valve (SRV) or Depressurization Valve (DPV) discharge, containment and suppression pool pressures and temperatures shall be maintained below their design values.

Based on the above, Table 15.0-3 lists the DCD Chapter 15 safety analysis acceptance criteria for AOO. Except for event-specific differences, Table 15.0-4 lists the DCD Chapter 15 safety analysis acceptance criteria for AOOs in combination with an additional SACF or SOE. These sets of acceptance criteria assume that all related safety analyses are performed with accepted models.

15.0.3.2 Infrequent Events

The ESBWR is designed such that any infrequent event would not result in the reactor water level dropping to below the top of the core (i.e., active fuel).

For a new plant, the 10 CFR regulations associate the consequences of postulated accidents with the exposures in 10 CFR 50.34(a)(1). Infrequent events do not result in a larger consequence than the least severe of the *DBAs*, and thus, their maximum radiological acceptance criteria should be ≤ 2.5 rem TEDE. However, if the SRP specifies a different or additional radiological acceptance criterion (e.g., a 10 CFR 20 limit or a different TEDE value), then the SRP acceptance criteria apply.

Based on the 10 CFR regulations and the SRP, GDC 19 is the only basis for the acceptance criterion on control room doses for all non-AOO abnormal event evaluations, such as infrequent events and accidents.

Based on ASME code classification of events with their associated stress limits and historical accepted criterion, infrequent events most closely correlate with ASME Code Service Level C limits. Therefore, reactor coolant system pressure should be based on the ASME Code Service Level C limit, which is conservatively interpreted to correspond to 120% of design pressure.

If an infrequent event results in an SRV/DPV discharge or fission product release to the containment, then containment stresses (i.e., pressures and temperatures) should be limited such that there is no loss of a containment barrier safety function, and thus, the containment must remain within its design limits/values.

Except for event-specific differences, Table 15.0-5 provides a generic set of acceptance criteria for infrequent event safety analyses.

15.0.3.3 Accidents

For a new plant, the 10 CFR regulations associate the consequences of postulated accidents with the exposures in 10 CFR 50.34(a)(1). Non-DBA accidents should not result in a larger consequence than the least severe of the DBAs, and thus, their radiological acceptance criteria should usually be limited to 2.5 rem TEDE. However, (like infrequent events) if the applicable SRP specifies a different or additional radiological acceptance criterion (e.g., a 10 CFR 20 limit or a different TEDE value), then the SRP acceptance criterion applies.

Based on the 10 CFR regulations and the SRP, GDC 19 is the only basis for the acceptance criterion on control room doses for all postulated accidents.

For the DBAs, the SRP 15.0.1 and RG 1.183 provide the consequence acceptance criteria of 2.5 rem TEDE, 6.3 rem TEDE and 25 rem TEDE [equivalent to 10%, 25% and 100% of the 10 CFR 50.34(a)(1) exposures], depending on the specific DBA. For DBAs, which do not have a consequence acceptance criterion specified in SRP 15.0.1 and RG 1.183, the smallest (i.e., 2.5 rem TEDE) criterion is applied.

For any accident that involves ECCS activation, the 10 CFR 50.46(a)(3)(b) acceptance criteria apply, and thus, the calculated changes in core geometry shall be such that the core remains amenable to cooling.

RG 1.70 classifies accidents as “limiting faults,” which can be correlated to different service levels or design conditions in the applicable industry code, e.g., ASME Code Service Level C or D. To ensure conservatism and minimize the number of acceptance condition options, for DBAs, reactor coolant pressure boundary components shall be limited to ASME Code Service Level C limits.

If an accident results in an SRV/DPV discharge or fission product release to the containment, then containment stresses (i.e., pressures and temperatures) should be limited such that there is no loss of a containment barrier safety function, and thus, the containment must remain within its design limits/values.

The set of acceptance criteria for accident safety analyses is provided in Table 15.0-6.

Because radiological acceptance criteria vary for the different event scenarios, for each non-AOO design basis event scenario applicable to an ESBWR, Table 15.0-7 provides radiological acceptance criteria.

15.0.3.4 Special Events

The acceptance criteria for each of the special event safety analyses is developed on an event-specific basis, based on the coping, mitigation or acceptance criteria specified in the 10 CFR regulations, the SRP or an NRC reviewed LTR.

15.0.3.4.1 MSIV Closure With Flux Scram

For every fuel cycle, a MSIV Closure With Flux Scram analysis (commonly referred to as the Overpressure Protection Analysis) is performed. With respect to the reactor coolant pressure boundary (RCPB) pressure response, the event scenario is specifically chosen to bound all of the design basis events.

The event requires/assumes:

- an operator error, multiple equipment failures or a common mode failure cause(s) the MSIVs in all four main steamlines (MSLs) to simultaneously close;
- the two MSIV position switch circuits on three to six MSIVs fail, which causes the MSIV position scram function to fail; and
- the reactor is shutdown by a high neutron flux scram trip.

The MSIV Closure With Flux Scram analysis demonstrates that the SRVs have adequate pressure relief capacity to prevent the RCPB ASME Code Service Level B pressure limit(s) and the Reactor Coolant System Pressure Safety Limit in the Technical Specifications from being exceeded.

Therefore, this event only needs/has the following acceptance criteria:

- Pressures in the reactor coolant and main steam systems shall be maintained below 110% of their design values (i.e., not exceed ASME Code Service Level B).
- The reactor steam dome pressure shall be maintained less than or equal to the Reactor Coolant System Pressure Safety Limit in the Technical Specifications.

15.0.3.4.2 Shutdown Without Control Rods

Assuming all control rod insertion mechanisms fail, for every fuel cycle, cold shutdown core k-effective (k_{eff}) calculations are performed at various cycle exposure points, to ensure that the Standby Liquid Control System (SLCS) can inject adequate (boron solution) negative reactivity into the core to allow for cold shutdown. This analysis plus the normal control rod shutdown margin calculations demonstrate compliance to GDC 26.

The Shutdown Without Control Rods event only needs/has the following acceptance criterion:

- Under the most reactive core conditions, k_{eff} shall be < 1.0.

15.0.3.4.3 Shutdown from Outside Main Control Room

A Shutdown from Outside Main Control Room safety analysis shall demonstrate that the plant can achieve and maintain safe shutdown, assuming the reactor is scrammed by the operators before they vacate the main control room.

The ability to cope with a Shutdown from Outside Main Control Room event is based on meeting the following acceptance criteria:

- Adequate reactor coolant inventory is maintained such that reactor water level is maintained above the core (i.e. top of active fuel).
- Achieve and maintain the plant to those shutdown conditions specified in plant Technical Specifications as Hot Shutdown.
- If containment isolation is involved, containment and suppression pool pressures and temperatures shall be maintained below their design values.

15.0.3.4.4 Anticipated Transient Without Scram (ATWS)

As documented in Reference 15.0-4, the generic BWR ATWS performance analysis acceptance criteria are summarized below.

- Pressures in the reactor coolant and main steam systems shall be maintained below ASME Service Level C limit, which is conservatively interpreted to correspond to 120% of design pressure.
- Peak cladding temperature within the 10 CFR 50.46 limit of 2200°F.
- Peak cladding oxidation within the requirements of 10 CFR 50.46.
- Peak suppression pool temperature shall not exceed its design temperature.
- Peak containment pressure shall not exceed containment design pressure.

15.0.3.4.5 Station Blackout (SBO)

An SBO safety analysis shall demonstrate that the plant can cope with the effects (i.e., with minimum equipment available) of an SBO for the duration of the SBO. The ability to cope with an SBO is based on meeting the following acceptance criteria.

- Adequate reactor coolant inventory is maintained such that reactor water level is maintained above the core (i.e. top of active fuel).
- Achieve and maintain the plant to those shutdown conditions specified in plant Technical Specifications as Hot Shutdown.
- If containment isolation is involved, containment and suppression pool pressures and temperatures shall be maintained below their design values.

15.0.3.4.6 Safe Shutdown Fire

The following acceptance criteria are derived from 10 CFR Part 50.48 and Appendix R.

- Core subcriticality is achieved and maintained with adequate core shutdown margin, as specified in the plant Technical Specifications.
- Adequate reactor coolant inventory is maintained such that reactor water level is maintained above the core (i.e., top of active fuel).

- Hot shutdown conditions are achieved and maintained.
- Cold shutdown conditions are achieved within 72 hours.
- Cold shutdown conditions are maintained thereafter.
- If containment isolation is involved, containment and suppression pool pressures and temperatures shall be maintained below their design values.
- Safety-related and nonsafety-related equipment may be used to meet the above criteria.

15.0.3.4.7 Waste Gas System Leak or Failure

Because the ESBWR Offgas System pressure boundary is designed to withstand dynamic overpressure from potential hydrogen detonation of at least 17 times the normal system operating pressure, a structural failure in the Offgas System is not a credible event. For the ESBWR, the only plausible event scenario that could result in a waste gas release requires two independent operator errors and an instrumentation isolation trip or (mechanical) isolation function failure to occur, and would result in only the release of noble gases. The postulation of a Waste Gas System Failure for the ESBWR goes beyond the 10 CFR 50 Appendix A single failure criterion, and thus, it does not qualify as a design basis event. This conclusion is consistent with SRP 15.7.1, which no longer requires this event to be analyzed within Chapter 15. Therefore, the Waste Gas System Failure for the ESBWR is classified as a special event.

The radiological analysis acceptance criterion for Waste Gas System Failure has had a number of different published values, as listed below.

The current NRC approved (1981) version of Branch Technical Position ETSB 11-5 has a 10 CFR 20 based acceptance criteria of 0.5 rem total whole body exposure. However, ETSB 11-5 does not state the specific paragraph within 10 CFR 20 from which the 0.5 rem value was taken.

Draft Rev. 3 - April 1996 version of ETSB 11-5, applies “a small fraction of 10 CFR 100 limit” (i.e., 2.5 rem total whole body) as the dose acceptance criterion.

The ABWR FSER NUREG-1503 Subsection 11.3.2 (page 11-11) gives the acceptance criterion for an “offgas system leak or failure as assumed in BTP ETSB 11-5, Revision 0, July 1981” as a whole body dose “less than 10 percent of the 10 CFR Part 100 limits.”

The NRC FSER for the AP1000 states “The BTP stipulates that the total body dose at the exclusion area boundary (EAB), as a result of the release of radioactivity for two hours from a postulated failure of the WGS, calculated in accordance with BTP assumptions, should not exceed 0.5 rem…. The applicant calculated a 0- to 2-hour total body dose within 0.5 rem, which satisfies BTP ETSB 11-5. Based on the above, the staff finds the analysis acceptable.”

In an RAI to Revision 1 of this report the NRC specified that the acceptance criterion is 0.1 rem TEDE, based on the 10 CFR 20.1301(a)(i), which states “ The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 mSv) in a year.”

During a July 26, 2005 NRC-GE conference call, the NRC stated that the use of the annual average atmospheric dispersion factor would be acceptable for a Waste Gas System Leak or Failure analysis that uses the 0.1 rem TEDE acceptance criterion. GE accepted this NRC position.

15.0.3.4.8 Potential Special Events

The 10 CFR regulations and the SRP do not contain a generic set of safety analysis acceptance criteria for special events. The safety analysis acceptance criteria for these events are on an event-specific basis. It is expected that any (potential) future special event will also have event-specific safety analysis acceptance criteria.

15.0.4 Event Analysis Format

For each event, the following information is provided in Sections 15.2, 15.3, 15.4 and 15.5.

15.0.4.1 Identification of Causes

Situations that lead to the analyzed events are described in their associated event descriptions. The frequency of occurrence of each event is summarized based upon the NSOA, currently available operating plant history for the abnormal event, and the evaluations in Appendix 15A. Events for which inconclusive data exist are discussed separately within each event section.

15.0.4.2 Sequence of Events and Systems Operations

Each event evaluated is discussed and evaluated in terms of:

- A step-by-step sequence of events from initiation to final stabilized condition.
- The extent to which normally operating plant instrumentation controls are assumed to function.
- The extent to which the plant and reactor protection systems are required to function.
- The credit taken for the functioning of normally operating plant systems.
- The operation of engineered safety systems that is required.

Each event's sequence of events is supported by the NSOA. The effect of a single equipment failure or malfunction or an operator error on the event is shown in the NSOA.

15.0.4.3 Evaluation of Results

- The results of the design basis events analyses are presented in Sections 15.2, 15.3 and 15.4. The limiting events can be identified, based on those results. Reasons why the other events are not limiting are given in the event documentation.

For the core loading in Figure 4.3-1, a representative MCPR operating limit is determined**.** Results of the AOO analyses for individual plant-specific core loading patterns will differ slightly from the results shown in this chapter. However, the relative results between core associated events do not change. The MCPR operating limit, for the as-built initial core and each reload core fuel loading pattern, will be provided by the COL licensee to the USNRC for information.

15.0.4.4 Barrier Performance

The significant areas of interest for internal pressure damage are the high-pressure portions of the RCPB (i.e., the reactor vessel and the high pressure pipelines attached to the reactor vessel).

15.0.4.5 Radiological Consequences

This subsection describes the consequences of radioactivity releases for the core loading, during DBEs. For all events whose consequences are limiting, a detailed quantitative evaluation is presented. For non-limiting events, a qualitative evaluation is presented or results are referenced from a more limiting or enveloping case or event.

15.0.5 Single Failure Criterion

From 10 CFR 50, Appendix A: "A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither

- a single failure of any active component (assuming passive components function properly) nor
- a single failure of a passive component (assuming active components function properly),

results in a loss of the capability of the system to perform its safety function. Single failures of passive components in electric systems should be assumed in designing against a single failure."

The single failure criterion (SFC) requires the plant design to be capable of providing specific functions during any design basis event (DBE) assuming a single failure in addition to the event initiating occurrence and any other coincident failures specified in the required DBE analysis assumptions. The application of the SFC to:

- The total plant is described in ANSI/ANS 52.1;
- Fluid systems are described in ANSI/ANS 58.9; and
- Electrical items are described in IEEE 379.

The IEEE criteria specify that electrical systems be designed to accommodate either a passive or an active single failure. For fluid systems in DBE analyses, the SFC only applies to active failures. The SFC is applicable to:

- Emergency core reactivity control (scram);
- Emergency core cooling;
- Reactor coolant pressure boundary isolation;
- Reactor coolant system pressure relief;
- Containment cooling;
- Containment isolation;
- Containment atmosphere clean up; and
- Their required supporting functions such as cooling water and electrical power.

Only one failure needs to be assumed per plant DBE, however, if a single occurrence can cause multiple failures, these multiple failures are treated as a single failure.

This subsection describes the application of single failure relative to DBEs. Single failure is defined in 10 CFR 50, Appendix A, and is specifically applied to multiple GDCs.

The treatment of plant capability evaluation events (i.e., special events) is consistent with their specific event definitions that are typically beyond the safety design bases of the plant. As a result, an additional single failure is not applied unless there is a specific licensing commitment.

15.0.5.1 Single Failures as Event Initiators

The AOOs identified in the safety analysis are frequently associated with transients that result from a single component failure or operator error, and are postulated during specific, applicable mode(s) of normal plant operation. Operator error is usually only considered as an event initiator.

Operator error is defined as a deviation from written operating procedures or operating practices. An operator error includes action(s) that are a direct consequence of one operator's single erroneous decision. An operator error does not include subsequent actions performed in response to the initiating event that resulted from the initial operator error.

Operator errors include:

- Erroneous selection and withdrawal of a control rod or control rod group.
- The manual isolation of the main steam lines caused by operator misinterpretation of an alarm or indicator.

15.0.5.2 Application of Single Failure Criteria to Event Analysis

The single-failure requirements for DBEs in the safety analysis and the NSOA are applied as follows:

- For DBEs, the protection sequences within mitigation systems are to be single-component-failure-proof. This position is in addition to any single-component failure or single operator error that is assumed as the event initiator. The requirement for assuming an additional single failure in the mitigation system adds a significant level of conservatism to the safety analysis. However, the event acceptance limits for DBEs are not changed by the application of an additional single-failure requirement.
- For AOOs, it is not always necessary to assume a single failure in normal operating systems in addition to the failure assumed as the event initiator. The basic logic for this assumption is based upon the probability of occurrence of a double failure in normal operating systems, which may be less than once per plant lifetime and exceeds the probability of occurrence definition for AOOs in 10 CFR 50, Appendix A.
- For infrequent events and accidents, single failures are considered consistent with plant-specific licensing commitments (e.g., valve malfunctions for LOCA).
- Multiple (consequential) failures from a single failure are considered part of the single failure. Single failures are independently postulated in each operating unit or one failure is postulated in the common systems.

- For mitigation systems included in the NSOA, single failures of active electrical and fluid components, and passive electrical components are treated in the same manner in the development of the event diagrams.
- During Technical Specifications surveillance testing or when complying with the limiting conditions for operation, applying the single-failure criteria for affected components/systems is not required. This is consistent with component/system reliability assumptions that form the bases for the plant-specific Technical Specifications.

The single failures identified above are considered in the design of the plant, as required by specific GDC, and are utilized in the safety analysis of the specific events.

15.0.6 References

15.0-1 GE Nuclear Energy, "Classification of ESBWR Abnormal Events and Determination of Their Safety Analysis Acceptance Criteria," NEDO-33175, Revision 2, E&A 2, August 2005.

15.0-2 GE Nuclear Energy, "GESTAR II General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (GE Proprietary) and NEDO-24011 (non-proprietary), latest revision.

15.0-3 USNRC, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design," NUREG-1503, Volume 1, July 1994.

15.0-4 General Electric Company, "Assessment of BWR Mitigation of ATWS, Volume II (NUREG 0460 Alternate No. 3)." NEDE-24222, Class III (proprietary), December 1979, and NEDO-24222, Class I (non-proprietary), February 1981.

Table 15.0-1

Chapter 15 Abnormal Event Classification Determination Matrix

Determination Criteria vs. Event Classification	Annual Probability $\geq 10^{-2}$	Thermal Hydraulic Basis	Radiological Analysis Basis		Assumes An Additional SACF or SOE		Event Not Included As A Design Basis Event in 10 CFR 50.49(b)(1)(ii) **and**		
			10 CFR 20	10 CFR 50.34(a)(1) & GDC 19	Yes	No	Is Postulated In A Regulation	Assumes Common-Mode Failure(s)	Assumes Failures, Beyond SFC
AOO	X	Greater than 99.9% of the fuel rods in the core would be expected to avoid boiling transition safety limit	(Not needed)			X			
		Maintain 100% Core Coverage	X		X				
Infrequent Event		Maintain 100% Core Coverage	X*	X*	X*	X*			
Accident		10 CFR 50.46		X	X				
Special Event **		X*	***	X***			X * +	X * +	X * +

*	Specific event dependent.

**	Does not include severe accidents and other events that are only evaluated as part of the plant PRA.***	If applicable to a specific special event.

\+	Or any combination of these conditions.

Table 15.0-2

ESBWR Abnormal Event Classifications

Abnormal Event	Event Classification	Relevant SRP(s)
Loss of Feedwater Heating	AOO	15.1.1 - 4
Closure of One Turbine Control Valve	AOO	15.2.1 – 5
Generator Load Rejection with Bypass	AOO	15.2.1 – 5
Generator Load Rejection with a Single Failure in the Turbine Bypass System	AOO	15.2.1 – 5
Turbine Trip with Bypass	AOO	15.2.1 – 5
Turbine Trip with a Single Failure in the Turbine Bypass System	AOO	15.2.1 – 5
Closure of One Main Steam Isolation Valve	AOO	15.2.1 – 5
Closure of All Main Steam Isolation Valves	AOO	15.2.1 – 5
Loss of Condenser Vacuum	AOO	15.2.1 – 5
Loss of Shutdown Cooling Function of RWCU/SDC	AOO	15.2.1 – 5
Inadvertent Isolation Condenser Initiation	AOO	15.1.1 – 4
Runout of One Feedwater Pump	AOO	15.1.1 – 4
Opening of One Turbine Control or Bypass Valve	AOO	15.1.1 – 4
Loss of Unit Auxiliary Transformer *	AOO	15.2.6
Loss of Grid Connection *	AOO	15.2.6
Loss of All Feedwater Flow	AOO	15.2.7
Loss of Feedwater Heating With Failure of Selected Control Rod Run-In	Infrequent Event	15.1.1 – 4
Feedwater Controller Failure – Maximum Demand	Infrequent Event	15.1.1 – 4
Pressure Regulator Failure - Opening of All Turbine Control and Bypass Valves	Infrequent Event	15.1.1 – 4
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves	Infrequent Event	15.1.1 – 4
Generator Load Rejection with Total Turbine Bypass Failure	Infrequent Event	15.2.1-5
Turbine Trip with Total Turbine Bypass Failure	Infrequent Event	15.2.1-5
Control Rod Withdrawal Error During Refueling	Infrequent Event	15.4.1

Table 15.0-2

ESBWR Abnormal Event Classifications

Abnormal Event	Event Classification	Relevant SRP(s)
Control Rod Withdrawal Error During Startup	Infrequent Event	15.4.1
Control Rod Withdrawal Error During Power Operation	Infrequent Event	15.4.2
Fuel Assembly Loading Error, Mislocated Bundle	Infrequent Event	15.4.7
Fuel Assembly Loading Error, Misoriented Bundle	Infrequent Event	15.4.7
Inadvertent SDC Function Operation	Infrequent Event	15.4.9
Inadvertent Opening of a Safety/Relief Valve	Infrequent Event	15.6.1
Inadvertent Opening of a Depressurization Valve	Infrequent Event	15.6.1, 15.6.5
Stuck Open Safety/Relief Valve	Infrequent Event	15.6.1
Liquid-Containing Tank Failure (COL applicant scope)	Infrequent Event	15.7.3
Spent Fuel Cask Drop Accident	Accident	15.7.5
Fuel Handling Accident	Accident	15.7.4
LOCA Inside Containment	Accident	15.6.5 & 5a
Main Steamline Break Outside Containment	Accident	15.6.4
Control Rod Drop Accident	Accident	15.4.9
Feedwater Line Break Outside Containment	Accident	15.3.5
Failure of Small Line Carrying Primary Coolant Outside Containment	Accident	15.6.2
RWCU/SDC System Line Failure Outside Containment	Accident	15.6.4, 15.6.5
MSIV Closure With Flux Scram (Overpressure Protection)	Special Event	5.2.2
Shutdown Without Control Rods (i.e., SLC system shutdown capability)	Special Event	9.3.5
Shutdown from Outside Main Control Room	Special Event	7.5
Anticipated Transients Without Scram	Special Event	15.8
Station Blackout	Special Event	8.2 (and RG 1.155)
Safe Shutdown Fire	Special Event	9.5.1
Waste Gas System Leak or Failure	Special Event	11.3

* Both covered by the Loss of Non-Emergency AC Power to Station Auxiliaries event.

Table 15.0-3

Safety Analysis Acceptance Criteria for AOOs

- Pressures in the reactor coolant and main steam systems shall be maintained below 110% of their design values (i.e., not exceed ASME Code Service Level B), and the reactor steam dome pressure shall be maintained less than or equal to the Reactor Coolant System Pressure Safety Limit in the Technical Specifications.
- Fuel-cladding integrity should be maintained by ensuring that the reactor core is designed with appropriate margin during any conditions of normal operation, including the effects of AOOs. The minimum value of the critical power ratio (CPR) reached during the AOO should be such that 99.9 percent of the fuel rods in the core would not be expected to experience boiling transition during core-wide transients. (This criterion corresponds to the <u>greater than 99.9% of the fuel rods in the core would be expected to avoid boiling transition related safety limit</u> in the Technical Specifications.)
- Uniform cladding strain ≤ 1%*.
- No fuel centerline melt (core-wide AOOs only).
- Energy generation is < 170 cal/g (RWE during startup only).
- Containment and suppression pool pressures and temperatures shall be maintained below their design values.
- An AOO should not generate a more serious plant condition unless other faults occur independently.
- There is no loss of function of any fission product barrier (Safety/Relief Valve or Depressurization Valve discharge does not apply).

* Based on SRP Sections 15.4.1 and 15.4.2, for the Uncontrolled Control Rod Assembly Withdrawal From a Subcritical or Low Power Startup Condition (i.e., control rod withdrawal error [RWE] during startup) event and the Uncontrolled Control Rod Assembly Withdrawal At Power (i.e., RWE during power operation) event.

Table 15.0-4

Safety Analysis Acceptance Criteria for AOOs In Combination With An Additional Single Active Component Failure or Single Operator Error

- Reactor water level shall be maintained above the top of the core (i.e., active fuel).
- Pressures in the reactor coolant and main steam systems shall be maintained below 110% of their design values (i.e, not exceed ASME Code Service Level B), and the reactor steam dome pressure shall be maintained less than or equal to the Reactor Coolant System Pressure Safety Limit in the Technical Specifications.
- If containment isolation is involved, containment and suppression pool pressures and temperatures shall be maintained below their design values.
- Except for fuel cladding, there shall be no loss of function of any fission product barrier.
- Fuel cladding failures shall be limited such that the radiological consequence shall be ≤ 0.05 rem TEDE.

Table 15.0-5

Safety Analysis Acceptance Criteria for Infrequent Events

- Reactor water level shall be maintained above the top of the core (i.e., active fuel).
- Pressures in the reactor coolant and main steam systems shall be maintained below the ASME Service Level C limit, which corresponds to 120% of design pressure.
- Radiological consequence shall be ≤ 2.5 rem TEDE. However, if the applicable SRP section specifies an accident-specific (i.e., different or additional) radiological acceptance criterion, then the accident-specific SRP acceptance criterion/criteria is/are applied. *
- Containment and suppression pool pressures and temperatures shall be maintained below their design values.
- Control room personnel shall not receive a radiation exposure in excess of 5 rem TEDE for the duration of the event.

* For example, the liquid radwaste tank failure must meet 10 CFR 20, Table 2, Column 2 for the liquid release.

Table 15.0-6

Safety Analysis Acceptance Criteria for Accidents

- Pressures in the reactor coolant and main steam systems shall be maintained below the ASME Service Level C limit, which corresponds to 120% of design pressure.
- Radiological consequence shall be ≤ 2.5 rem TEDE, 6.3 rem TEDE, or 25 rem TEDE, depending on the accident-specific acceptance criterion in NUREG-0800, SRP 15.0.1.
- The calculated maximum fuel element cladding temperature shall not exceed 1204ºC (2200ºF).
- The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
- Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.
- Containment and suppression pool pressures and temperatures shall be maintained below their design values.
- Control room personnel shall not receive a radiation exposure in excess of 5 rem TEDE for the duration of the accident.

Table 15.0-7

ESBWR Event Classifications and Radiological Acceptance Criteria

Event*	Accident Class**		Radiological Acceptance Criteria***					
	Infrequent Event	Accident	10 CFR 20, App. B, Table 2, Column 2	10 CFR 20.1301	GDC 19, 5 rem TEDE	2.5 rem TEDE	6.3 rem TEDE	25 rem TEDE
Loss of Feedwater Heating With Failure of Selected Control Rod Run-In	X				+	X		
Inadvertent SDC Function Operation	X				+	X		
Control Rod Withdrawal Error During Refueling	X				+	X		
Control Rod Withdrawal Error During Startup	X				+	X		
Control Rod Withdrawal Error During Power Operation	X				+	X		
Inadvertent Opening of a Depressurization Valve	X				+	X		
Inadvertent Opening of a Safety/Relief Valve	X				+	X		
Stuck Open Safety/Relief Valve	X				+	X		
Feedwater Controller Failure – Maximum Demand	X				+	X		
Pressure Regulator Failure - Opening of All Turbine Control and Bypass Valves	X				+	X		
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves	X				+	X		
Generator Load Rejection with Total Turbine Bypass Failure	X				+	X		
Turbine Trip with Total Turbine Bypass Failure	X				+	X		
Liquid-Containing Tank Failure	X		X		+			

Table 15.0-7

ESBWR Event Classifications and Radiological Acceptance Criteria

Event*	Accident Class**		Radiological Acceptance Criteria***					
	Infrequent Event	Accident	10 CFR 20, App. B, Table 2, Column 2	10 CFR 20.1301	GDC 19, 5 rem TEDE	2.5 rem TEDE	6.3 rem TEDE	25 rem TEDE
Fuel Assembly Loading Errors (mislocated and misoriented)	X				+	X		
Spent Fuel Cask Drop Accident		X			+		X	
Failure of Small Line Carrying Primary Coolant Outside Containment		X			+	X		
Feedwater Line Break Outside Containment		X			+	X		
Reactor Water Cleanup / Shutdown Cooling System Failure Outside Containment		X			+	X		
Control Rod Drop Accident (radiological analysis)		X			+		X	
Main Steamline Break Outside Containment		X			X	X		
LOCA Inside Containment Radiological Analysis, (including all leakage paths)		X			X			X
Fuel Handling Accident		X			+		X	
Waste Gas System Leak or Failure ++				X	+			

* Based on SRP 15 and ABWR FSER events involving a radiological consequence.

** From Table 15.0-2.

*** Based on the 10 CFR regulations and SRP 15.

\+ Bounded by the LOCA Inside Containment and Main Steamline Break Outside Containment radiological analyses.

++ Classified as a special event.

15.1 NUCLEAR SAFETY OPERATIONAL ANALYSIS

The Nuclear Safety Operational Analysis (NSOA) is a system level qualitative failure modes and effects analysis (FMEA) of plant protective functions that shows which systems and functions are required for the events addressed in Chapter 15 to meet their associated acceptance criteria.

15.1.1 Analytical Approach

15.1.1.1 NSOA Objective

The objective of the NSOA is to identify, for each event in the Chapter 15 safety analyses, the system level requirements that ensure the plant can be brought to a stable safe condition. Specifically, the NSOA considers the entire duration of each event from the spectrum of possible initial conditions and aftermath until either some mode of planned operation is resumed or the plant is in a safe shutdown condition.

The NSOA process uses operational criteria and required actions to identify the required systems, automatic instrument trips, monitored parameters (associated with required operator actions), and auxiliary systems to bring the plant to a stable condition for each event. The system-level requirements identified as required in the NSOA reflect the licensing basis of the plant and constitute the minimum required actions to bring the plant to a stable safe condition. In actual plant operation, additional procedural guidance and plant equipment are available to prevent or further mitigate these events. Finally, the NSOA focuses primarily on active plant features used to bring the plant to a stable safe condition; passive plant features are implicitly considered but not explicitly documented in the event evaluations and diagrams.

15.1.1.2 NSOA Relationship to Safety Analysis

The safety analysis is performed to demonstrate compliance with appropriate event acceptance criteria (Subsection 15.0.2) for limiting event paths. Review of the event acceptance criteria illustrates the safety analysis focus on event consequences. The event acceptance criteria are either fission product barrier design basis limits or radiological dose limits derived from applicable regulatory requirements.

As such, the event paths analyzed as "limiting" in the safety analysis generally correspond to one of the event paths for each event in the NSOA, or a conservative representation of one.

This safety analysis limiting-event path is selected to pose the most significant challenge to the applicable event acceptance criteria, and thus, typically concentrates on the short-term response to the event. Therefore, the safety analysis is consequences oriented, focusing on the limiting short-term response to the event, and the NSOA is event/system oriented, focusing on the system-level required actions necessary over the entire duration of the event (long-term response) to bring the plant to a stable configuration.

15.1.2 Method of Analysis

15.1.2.1 Operational Criteria

The operational criteria are identified in Table 15.1-1.

The operational criteria establish the requirements for:

- Satisfying the applicable required actions to bring the plant to a stable condition consistent with the plant licensing basis.
- Applying the single failure criteria
- Satisfying requirements unique to certain events.

Operational criteria are based upon the applicable regulatory requirements and guidance, industry codes and standards, plant-specific licensing requirements, nuclear steam supply system (NSSS) requirements, and fuel supplier design requirements.

15.1.2.2 Analysis Assumptions and Initial Conditions

15.1.2.2.1 Operating States

Four boiling water reactor (BWR) operating states encompassing the entire operating envelope in which the plant can exist are defined in Table 15.1-2. The main objective in selecting operating states is to divide the plant operating spectrum into sets of initial conditions. This facilitates consideration of various events in each state. The events associated with each operating state are provided in Table 15.1-3.

Operating states are differentiated by a significant change in operational characteristics. The selection of not shutdown versus shutdown is based upon differences in reactivity control requirements. Requirements to shut down the reactor, under certain circumstances, are replaced by refueling interlock requirements. The selection of vented versus not vented is based upon differences in core cooling requirements. For example, when the reactor pressure vessel (RPV) is vented, the loss-of-coolant accident (LOCA) is not considered credible.

Each operating state includes an allowable range of values for important plant parameters. Within each state, these parameters are considered over their entire range.

For each event, the operating states in which the event can occur are determined. An event is considered applicable within an operating state if it can be initiated from the operating envelope and operating modes that characterize the operating state.

15.1.2.2.2 Operating Modes

The operating states encompass all operating modes associated with planned operation and their respective operating envelopes. The plant operating modes, associated with each operating state, are identified in Table 15.1-2.

Together, the BWR operating states and the operating modes associated with planned operations define the operating envelope from which anticipated operational occurrences (AOOs), Infrequent Events, Accidents, and Special Events are initiated. ESBWR operating states define the physical condition (e.g., pressure, temperature) of the reactor. Operating modes define what the plant is doing. The separation of the physical conditions from the operation being performed is deliberate and facilitates careful consideration of all possible initial conditions from which events may be postulated to occur.

15.1.2.2.3 Planned Operation

Planned operation refers to normal plant operation under planned conditions within the allowable operating envelope in the absence of significant abnormalities. Following an event, planned operation is not considered to have resumed until the plant operating state is identical to a planned operating mode that could have been attained had the event not occurred. As defined, planned operation can be considered as a chronological sequence:

Refueling outage > achieving criticality > heatup > power operation > achieving shutdown > cooldown > refueling outage

15.1.2.3 Event Analysis Rules

The event analysis rules are consistent with applicable regulatory requirements and guidance, plant-specific licensing commitments, and applicable industry codes and standards. Table 15.1-4 provides the event analysis rules used in performing the NSOA, along with explanations of the individual rules.

15.1.3 NSOA Results

15.1.3.1 Event Evaluations and Diagrams

The individual event evaluations in conjunction with their respective event diagrams document the detailed results of the NSOA. The event diagram format is shown in Figure 15.1-1. The event evaluations are provided in Subsection 15.1.4 and the associated event diagrams are shown in Figures 15.1-3 through 15.1-49.

An event diagram for each event evaluated identifies the applicable operating states (for the overall event evaluation and, where applicable, for event paths that only apply to specific operating states), the required actions, the relationship of system operation and operator actions to the required actions, and the required functional redundancy. In addition, event diagrams identify each signal that initiates automatic system operation or alerts the operator to the need for action.

15.1.3.2 Auxiliary System Evaluation and Diagrams

Auxiliary systems are systems required for the proper functioning of front-line or other auxiliary systems. These systems are shown on auxiliary system diagrams. The auxiliary system diagram format is shown in Figure 15.1-2.

15.1.3.3 Summary Matrices

A system, instrument trip, or operator action is considered "required" if identified on an event diagram as necessary to satisfy a required action or the operational criteria. Required auxiliary systems for each event are identified via the auxiliary system diagrams.

Based upon the event evaluations and diagrams, matrices are provided in Table 15.1-5 and Table 15.1-6 to identify the required systems and automatic instrument trips, respectively for the events evaluated in the NSOA and the safety analyses.

15.1.4 Event Evaluations

The events considered in the NSOA are shown in Table 15.1-7 along with the locations of the event descriptions and the relevant protection sequence diagram.

Table 15.1-1

Operational Criteria

Applicability	Criteria
1. Planned operation	The plant shall be operated observing operating state monitoring requirements identified to preserve safety analysis assumptions and establish initial conditions for event analyses. Normal plant operating procedures are followed as applicable.
2. All events	All required actions to bring the plant to a stable condition consistent with the plant licensing basis shall be satisfied.
3. All events	Emergency Operating Procedures (EOPs) are followed when applicable.
4. AOOs	The plant shall be designed and operated such that no single failure in mitigation systems can prevent required actions from being satisfied.
5. Infrequent Events and Accidents	The plant shall be designed and operated to satisfy required actions, considering limiting single failure as defined by applicable regulatory requirements and licensing commitments.
6. AOOs, Infrequent Events and Accidents	Single-failure criterion is not applicable during periods of system or component testing required by Technical Specifications (TS) or when operating under limiting conditions for operation required by Technical Specifications.
7. Special events	The plant shall be designed and operated consistent with applicable regulatory requirements and licensing commitments.

Table 15.1-2

ESBWR Operating States/Operating Modes

STATE A – RPV VENTED AND REACTOR SHUTDOWN

- Allowable Mode Switch Positions: SHUTDOWN REFUEL
- Pressure Considerations: Atmospheric Pressure
- Power Considerations: Decay Heat Only

STATE B – RPV VENTED AND REACTOR NOT SHUTDOWN

- Allowable Mode Switch Positions: SHUTDOWN REFUEL STARTUP
- Pressure Considerations: Atmospheric Pressure
- Power Considerations: Decay Heat Only

STATE C – RPV HEAD ON (RPV NOT VENTED) AND REACTOR SHUTDOWN

- Allowable Mode Switch Positions: SHUTDOWN REFUEL STARTUP
- Pressure Considerations: Hot Shutdown ≥ Reactor Pressure ≥ Shutdown Cooling Permissive -or- Shutdown Cooling Permissive ≥ Reactor Pressure ≥ Atmospheric
- Power Considerations: Decay Heat Only

STATE D – RPV HEAD ON (RPV NOT VENTED) AND REACTOR NOT SHUTDOWN

- Allowable Mode Switch Positions: STARTUP RUN
- Pressure Considerations: Normal Operation ≥ Reactor Pressure ≥ Shutdown Cooling Permissive -or- Shutdown Cooling Permissive ≥ Reactor Pressure ≥ Atmospheric
- Power Considerations: Licensed Power Level ≥ Reactor Power ≥ Turbine Scram Bypass

Table 15.1-3

ESBWR Events Associated With Operating States

Abnormal Event	Applicable Operating State(s)
Loss of Feedwater Heating	D
Closure of One Turbine Control Valve	D
Generator Load Rejection with Turbine Bypass	D
Generator Load Rejection with a Single Failure in the Turbine Bypass System	D
Turbine Trip with Turbine Bypass	D
Turbine Trip with a Single Failure in the Turbine Bypass System	D
Closure of One Main Steam Isolation Valve	C, D
Closure of All Main Steam Isolation Valves	C, D
Loss of Condenser Vacuum	C, D
Loss of Shutdown Cooling Function of RWCU/SDC System	A, B, C, D
Inadvertent Isolation Condenser Initiation	D
Runout of One Feedwater Pump	D
Opening of One Turbine Control or Bypass Valve	C, D
Loss of Unit Auxiliary Transformer	A, B, C, D
Loss of Grid Connection	A, B, C, D
Loss of All Feedwater Flow	D
Loss of Feedwater Heating With Failure of Selected Control Rod Run-In	D
Feedwater Controller Failure – Maximum Demand	D
Pressure Regulator Failure Opening of All Turbine Control and Bypass Valves	C, D
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves	D
Generator Load Rejection with Total Turbine Bypass Failure	D
Turbine Trip with Total Turbine Bypass Failure	D
Control Rod Withdrawal Error During Refueling	A

Table 15.1-3

ESBWR Events Associated With Operating States

Abnormal Event	Applicable Operating State(s)
Control Rod Withdrawal Error During Startup	B, C
Control Rod Withdrawal Error During Power Operation	D
Fuel Assembly Loading Error, Mislocated Bundle	A, B, C, D
Fuel Assembly Loading Error, Misoriented Bundle	A, B, C, D
Inadvertent SDC Function Operation	A, B, C, D
Inadvertent Opening of a Safety/Relief Valve	C, D
Inadvertent Opening of a Depressurization Valve	C, D
Stuck Open Safety/Relief Valve	C, D
Liquid-Containing Tank Failure	A, B, C, D
Fuel Handling Accident	A
LOCA Inside Containment	C, D
Main Steamline Break Outside Containment	C, D
Control Rod Drop Accident	A, B, C, D
Feedwater Line Break Outside Containment	C, D
Failure of Small Line Carrying Primary Coolant Outside Containment	C, D
RWCU/SDC System Line Failure Outside Containment	C, D
Spent Fuel Cask Drop Accident	A, B, C, D
MSIV Closure With Flux Scram (Overpressure Protection)	D
Shutdown Without Control Rods (i.e., SLCS shutdown capability)	B, D
Shutdown from Outside Main Control Room	B, D
Anticipated Transients Without Scram	D
Station Blackout	D
Safe Shutdown Fire	D
Waste Gas System Leak or Failure	A, B, C, D

Table 15.1-4
Event Analysis Rules

A General Rules	Explanation
A.1 Include all events that are part of the plant safety analysis.	All events considered in the plant safety analysis are included in the NSOA, consistent with NSOA goals and objectives.
A.2 Identify on event diagrams all required systems, automatic trips, and operator actions necessary to either satisfy operational criteria or perform required actions.	Systems, automatic trips, and operator actions are identified only if they are uniquely necessary to either accomplish required actions or satisfy operational criteria.
A.3 Identify all support or auxiliary systems on auxiliary system diagrams.	Auxiliary systems are systems required to enable front-line systems (systems identified on event diagrams) or other auxiliary systems to perform their required functions.
A.4 Consider all plant systems, including passive plant features required in the mitigation of events.	The functions of passive plant features (e.g., MSL flow restrictors and CRD housing supports) used to mitigate the consequences of events are identified.
A.5 Consider hardware restrictions included in the plant design to prevent operation outside the operating envelope.	Hardware restrictions (e.g., control rod withdrawal restrictions and refueling interlocks) are included in the plant design to constrain plant operation to within the allowable operating envelope.
B Planned Operation Rules	**Explanation**
B.1 Consider only systems, limits, and restrictions necessary to attain planned operation and satisfy operational criteria.	Consideration of planned operation is limited and not followed through to completion, because planned operation is constrained by normal plant operating procedures.
B.2 Limit the initial conditions for AOOs, accidents, and special events to operating modes and envelopes allowed during planned operation in the applicable operating state.	All events in the safety analysis are initiated from an operating mode within the allowable operating envelope.
B.3 Consider the full range of initial conditions for each event analyzed.	This rule assures that all event paths are identified. Different initial conditions can lead to different paths that may establish unique requirements.

Table 15.1-4

Event Analysis Rules

B.4 Apply hardware restrictions only to planned operation.	Restrictions are hardware-implemented constraints on normal plant operation to limit the consequences of postulated events.
B.5 Identify normal operating systems considered for a planned operation function during an event as “Planned Operation - Specific System Available.”	Normal operating systems are considered if the system is employed in the same manner during the event as it was prior to the event or if continued operation can significantly change the event path.
C Event Diagram Rules	**Explanation**
C.1 Consider the entire duration of the event from the spectrum of possible initial conditions and aftermath until either some mode of planned operation is resumed or the plant is in a stable condition with continuity of core cooling.	Planned operation is considered "resumed" when normal operating procedures are being followed and plant operation is identical to that used in any operating state consistent with allowable operating modes and envelopes. A stable operating condition is defined as the completion of all required actions and the stabilization of plant parameters.
C.2 Identify systems, automatic trips, and operator actions if there is a unique requirement as a result of the event. If a normal operating system that was operating prior to the event will be employed in the same manner during the event and if the event did not affect system operation, the system does not appear as a unique requirement on the event diagram.	Systems, limits, and operator actions are identified as "required" only if a unique requirement to satisfy either required actions or operational criteria is established. When normal operating systems are considered, specific systems assumed to be available are identified.
C.3 Credit operator action only if the operator can reasonably be expected to accomplish the required action under existing conditions and has availability of necessary information to implement required plant procedures.	Operator action may be necessary to either attain planned operation or a stable condition.
C.4 Identify two types of parameters: Parameters that initiate an automatic trip or system actuation and monitored parameters (available to the operator) that require action.	Parameters are instrument setpoints at which either an automatic trip or system initiation or operator action is assumed to occur. Where either an automatic action or operator action accomplishes the same function, the automatic action is identified.

Table 15.1-4

Event Analysis Rules

C.5 Consider a system that plays a unique role in response to an AOO, accident, DBA or special event to be "required" unless the system's effects are not included in the event analysis.	Systems that have a unique role in an event are considered "required" unless the safety analysis for the event provides a basis that operation of the system is not required.
C.6 Identify operating states in which the event is applicable.	Because of plant operational considerations and the definition of operating states, not all events can occur in all operating states.
C.7 Identify the essential paths that include: • Required actions. • Front-line systems. • Automatic trips. • Monitored parameters. • Normal operating systems evaluated in analysis.	Event diagrams are the primary source of documentation of NSOA results. Notes identify required actions that are not applicable and required actions satisfied by the normal operating systems.
C.8 Identify passive plant features necessary at the system level.	Passive plant features are associated with system level requirements but are not included on the event diagrams, because they add unnecessary complexity.

Table 15.1-5

NSOA System Event Matrix

	SRV – Safety Relief Mode	SRV – Power Actuated Mode (ADS)	DPV Actuation	ICS – MSIV Closure	ICS – RPV High Dome Pressure	ICS – RPV Low Water Level (L2)	ICS – RPV Low Water Level (L1.5)	ICS – RPV Low Water Level (L1)	ICS – RPV Low Water Level (Loss of Offsite Power)	TBV Initiation – TSV Closure	TBV Initiation – TCV Fast Closure	TSV Closure – RPV High Water Level (L8)	TSV Closure – Low Condenser Vacuum	TCV Fast Closure – Load Rejection	MSIV Closure – RPV Low Water Level (L2 w/30 sec)	MSIV Closure – RPV Low Water Level (L1.5)	MSIV Closure – Low Turbine Inlet Pressure	MSIV Closure – Low Main Condenser Vacuum	FW Trip – RPV High Water Level (L9)	FW Pump Runback – L8	CRD Makeup Water – RPV Low Water Level (L2)	RWCU/SDC Operation	ATWS – ARI	ATWS – Electrical Insertion of FMCRDs	ATWS – Feedwater Flow Runback	ATWS – ADS Inhibition	SLCS – RPV Dome High Pressure	SLCS – DPV Open	SLCS – RPV Low Water Level (L2) w/High DW P	FAPCS – High Suppression Pool Temperature	SCRRIb	GDCS	GDCS Equilizing Lines	High Radiation MCR Recirculation
Loss of Feedwater Heating																															X			
Closure of One Turbine Control Valve																																		
Generator Load Rejection with Bypass											X			X																	X			
Generator Load Rejection with a Single Failure in the Turbine Bypass System						X					X			X							X													
Turbine Trip with Bypass										X																					X			
Turbine Trip with a Single Failure in the Turbine Bypass System						X				X											X													
Closure of One Main Steam Isolation Valve																																		
Closure of All Main Steam Isolation Valves				X																	X													
Loss of Condenser Vacuum				X									X					X			X													
Loss of Shutdown Cooling Function of RWCU/SDC System																																		

Table 15.1-5

NSOA System Event Matrix

	SRV – Safety Relief Mode	SRV – Power Actuated Mode (ADS)	DPV Actuation	ICS – MSIV Closure	ICS – RPV High Dome Pressure	ICS – RPV Low Water Level (L2)	ICS – RPV Low Water Level (L1.5)	ICS – RPV Low Water Level (L1)	ICS – RPV Low Water Level (Loss of Offsite Power)	TBV Initiation – TSV Closure	TBV Initiation – TCV Fast Closure	TSV Closure – RPV High Water Level (L8)	TSV Closure – Low Condenser Vacuum	TCV Fast Closure – Load Rejection	MSIV Closure – RPV Low Water Level (L2 w/30 sec)	MSIV Closure – RPV Low Water Level (L1.5)	MSIV Closure – Low Turbine Inlet Pressure	MSIV Closure – Low Main Condenser Vacuum	FW Trip – RPV High Water Level (L9)	FW Pump Runback – L8	CRD Makeup Water – RPV Low Water Level (L2)	RWCU/SDC Operation	ATWS – ARI	ATWS – Electrical Insertion of FMCRDs	ATWS – Feedwater Flow Runback	ATWS – ADS Inhibition	SLCS – RPV Dome High Pressure	SLCS – DPV Open	SLCS – RPV Low Water Level (L2) w/High DW P	FAPCS – High Suppression Pool Temperature	SCRRIb	GDCS	GDCS Equilizing Lines	High Radiation MCR Recirculation
Inadvertent Isolation Condenser Initiation																																		
Runout of One Feedwater Pump																																		
Opening of One Turbine Control or Bypass Valve																																		
Loss of Unit Auxiliary Transformer				X							X			X				X			X													
Loss of Grid Connection				X							X			X				X			X													
Loss of All Feedwater Flow				X											X						X													
Loss of Feedwater Heating With Failure of SCRRI																																		
Feedwater Controller Failure – Maximum Demand						X						X								X	X													
Pressure Regulator Failure Opening of All Turbine Control and Bypass Valves				X													X				X													

Table 15.1-5

NSOA System Event Matrix

	SRV – Safety Relief Mode	SRV – Power Actuated Mode (ADS)	DPV Actuation	ICS – MSIV Closure	ICS – RPV High Dome Pressure	ICS – RPV Low Water Level (L2)	ICS – RPV Low Water Level (L1.5)	ICS – RPV Low Water Level (L1)	ICS – RPV Low Water Level (Loss of Offsite Power)	TBV Initiation – TSV Closure	TBV Initiation – TCV Fast Closure	TSV Closure – RPV High Water Level (L8)	TSV Closure – Low Condenser Vacuum	TCV Fast Closure – Load Rejection	MSIV Closure – RPV Low Water Level (L2 w/30 sec)	MSIV Closure – RPV Low Water Level (L1.5)	MSIV Closure – Low Turbine Inlet Pressure	MSIV Closure – Low Main Condenser Vacuum	FW Trip – RPV High Water Level (L9)	FW Pump Runback – L8	CRD Makeup Water – RPV Low Water Level (L2)	RWCU/SDC Operation	ATWS – ARI	ATWS – Electrical Insertion of FMCRDs	ATWS – Feedwater Flow Runback	ATWS – ADS Inhibition	SLCS – RPV Dome High Pressure	SLCS – DPV Open	SLCS – RPV Low Water Level (L2) w/High DW P	FAPCS – High Suppression Pool Temperature	SCRRIb	GDCS	GDCS Equilizing Lines	High Radiation MCR Recirculation
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves						X															X													
Generator Load Rejection with Total Turbine Bypass Failure						X								X	X						X													
Turbine Trip with Total Turbine Bypass Failure						X									X						X													
Control Rod Withdrawal Error During Refueling																																		
Control Rod Withdrawal Error During Startup																																		
Control Rod Withdrawal Error During Power Operation																																		
Fuel Assembly Loading Error, Mislocated Bundle																																		
Fuel Assembly Loading Error, Misoriented Bundle																																		

Table 15.1-5

NSOA System Event Matrix

	SRV – Safety Relief Mode	SRV – Power Actuated Mode (ADS)	DPV Actuation	ICS – MSIV Closure	ICS – RPV High Dome Pressure	ICS – RPV Low Water Level (L2)	ICS – RPV Low Water Level (L1.5)	ICS – RPV Low Water Level (L1)	ICS – RPV Low Water Level (Loss of Offsite Power)	TBV Initiation – TSV Closure	TBV Initiation – TCV Fast Closure	TSV Closure – RPV High Water Level (L8)	TSV Closure – Low Condenser Vacuum	TCV Fast Closure – Load Rejection	MSIV Closure – RPV Low Water Level (L2 w/30 sec)	MSIV Closure – RPV Low Water Level (L1.5)	MSIV Closure – Low Turbine Inlet Pressure	MSIV Closure – Low Main Condenser Vacuum	FW Trip – RPV High Water Level (L9)	FW Pump Runback – L8	CRD Makeup Water – RPV Low Water Level (L2)	RWCU/SDC Operation	ATWS – ARI	ATWS – Electrical Insertion of FMCRDs	ATWS – Feedwater Flow Runback	ATWS – ADS Inhibition	SLCS – RPV Dome High Pressure	SLCS – DPV Open	SLCS – RPV Low Water Level (L2) w/High DW P	FAPCS – High Suppression Pool Temperature	SCRRIb	GDCS	GDCS Equilizing Lines	High Radiation MCR Recirculation
Inadvertent SDC Function Operation																																		
Inadvertent Opening of a Safety/Relief Valve																														X				
Inadvertent Opening of a DPV																														X				
Stuck Open Safety/Relief Valve						X															X									X				
Liquid-Containing Tank Failure																																		
Fuel Handling Accident																																		
LOCA Inside Containment		X	X			X	X	X	X						X	X					X							X	X			X	X	X
Main Steamline Break Outside Containment		X	X	X		X	X	X	X						X	X	X				X							X	X			X	X	X
Control Rod Drop Accident																																		
Feedwater Line Break Outside Containment		X	X	X	X	X	X	X	X						X	X					X							X	X			X	X	X
Failure of Small Line Carrying Primary Coolant Outside Containment		X	X	X	X	X	X	X	X						X	X					X							X	X			X	X	X

Table 15.1-5

NSOA System Event Matrix

	SRV – Safety Relief Mode	SRV – Power Actuated Mode (ADS)	DPV Actuation	ICS – MSIV Closure	ICS – RPV High Dome Pressure	ICS – RPV Low Water Level (L2)	ICS – RPV Low Water Level (L1.5)	ICS – RPV Low Water Level (L1)	ICS – RPV Low Water Level (Loss of Offsite Power)	TBV Initiation – TSV Closure	TBV Initiation – TCV Fast Closure	TSV Closure – RPV High Water Level (L8)	TSV Closure – Low Condenser Vacuum	TCV Fast Closure – Load Rejection	MSIV Closure – RPV Low Water Level (L2 w/30 sec)	MSIV Closure – RPV Low Water Level (L1.5)	MSIV Closure – Low Turbine Inlet Pressure	MSIV Closure – Low Main Condenser Vacuum	FW Trip – RPV High Water Level (L9)	FW Pump Runback – L8	CRD Makeup Water – RPV Low Water Level (L2)	RWCU/SDC Operation	ATWS – ARI	ATWS – Electrical Insertion of FMCRDs	ATWS – Feedwater Flow Runback	ATWS – ADS Inhibition	SLCS – RPV Dome High Pressure	SLCS – DPV Open	SLCS – RPV Low Water Level (L2) w/High DW P	FAPCS – High Suppression Pool Temperature	SCRRIb	GDCS	GDCS Equilizing Lines	High Radiation MCR Recirculation
RWCU/SDC System Line Failure Outside Containment		X	X	X	X	X	X	X	X						X	X					X							X	X			X	X	X
Spent Fuel Cask Drop Accident																																		
MSIV Closure With Flux Scram (Overpressure Protection)	X																																	
Shutdown Without Control Rods (i.e., SLCS shutdown capability)																																		
Shutdown from Outside Main Control Room				X											X						X	X								X				
Anticipated Transients Without Scram	X			X	X	X		X													X				X	X	X		X					
Station Blackout		X	X			X	X		X						X	X											X		X	X		X	X	
Safe Shutdown Fire				X											X						X	X								X				
Waste Gas System Leak or Failure																																		

Table 15.1-6

NSOA Automatic Instrument Trip/Event Matrix

	Scram – APRM High Neutron Flux	Scram – APRM High Simulated Thermal Power	Scram – RPV Low Water Level (L3)	Scram – RPV High Water Level (L8)	Scram – RPV High Dome Pressure	Scram – MSIV Closure	Scram – High Suppression Pool Temperature	Scram – TSV Closure (with TBP Inoperable)	Scram – TCV Fast Closure (with TBP Inoperable)	Scram – Insufficient Bypass Available	Scram – Low Condenser Vacuum	Scram – Feedwater not available	Scram – SRNM Period	Scram – High Drywell Pressure	Scram – Loss of two FW pumps (LOOP)	Rod Block – SRNM Period
Loss of Feedwater Heating																
Closure of One Turbine Control Valve																
Generator Load Rejection with Bypass																
Generator Load Rejection with a Single Failure in the Turbine Bypass System										X						
Turbine Trip with Bypass																
Turbine Trip with a Single Failure in the Turbine Bypass System										X						

Table 15.1-6

NSOA Automatic Instrument Trip/Event Matrix

	Scram – APRM High Neutron Flux	Scram – APRM High Simulated Thermal Power	Scram – RPV Low Water Level (L3)	Scram – RPV High Water Level (L8)	Scram – RPV High Dome Pressure	Scram – MSIV Closure	Scram – High Suppression Pool Temperature	Scram – TSV Closure (with TBP Inoperable)	Scram – TCV Fast Closure (with TBP Inoperable)	Scram – Insufficient Bypass Available	Scram – Low Condenser Vacuum	Scram – Feedwater not available	Scram – SRNM Period	Scram – High Drywell Pressure	Scram – Loss of two FW pumps (LOOP)	Rod Block – SRNM Period
Closure of One Main Steam Isolation Valve																
Closure of All Main Steam Isolation Valves						X										
Loss of Condenser Vacuum											X					
Loss of Shutdown Cooling Function of RWCU/SDC System																
Inadvertent Isolation Condenser Initiation																
Runout of One Feedwater Pump																
Opening of One Turbine Control or Bypass Valve																

Table 15.1-6

NSOA Automatic Instrument Trip/Event Matrix

	Scram – APRM High Neutron Flux	Scram – APRM High Simulated Thermal Power	Scram – RPV Low Water Level (L3)	Scram – RPV High Water Level (L8)	Scram – RPV High Dome Pressure	Scram – MSIV Closure	Scram – High Suppression Pool Temperature	Scram – TSV Closure (with TBP Inoperable)	Scram – TCV Fast Closure (with TBP Inoperable)	Scram – Insufficient Bypass Available	Scram – Low Condenser Vacuum	Scram – Feedwater not available	Scram – SRNM Period	Scram – High Drywell Pressure	Scram – Loss of two FW pumps (LOOP)	Rod Block – SRNM Period
Loss of Unit Auxiliary Transformer											X					
Loss of Grid Connection											X					
Loss of All Feedwater Flow												X				
Loss of Feedwater Heating With Failure of SCRRI		No credit														
Feedwater Controller Failure – Maximum Demand				X												
Pressure Regulator Failure Opening of All Turbine Control and Bypass Valves						X										

Table 15.1-6

NSOA Automatic Instrument Trip/Event Matrix

	Scram – APRM High Neutron Flux	Scram – APRM High Simulated Thermal Power	Scram – RPV Low Water Level (L3)	Scram – RPV High Water Level (L8)	Scram – RPV High Dome Pressure	Scram – MSIV Closure	Scram – High Suppression Pool Temperature	Scram – TSV Closure (with TBP Inoperable)	Scram – TCV Fast Closure (with TBP Inoperable)	Scram – Insufficient Bypass Available	Scram – Low Condenser Vacuum	Scram – Feedwater not available	Scram – SRNM Period	Scram – High Drywell Pressure	Scram – Loss of two FW pumps (LOOP)	Rod Block – SRNM Period
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves	X															
Generator Load Rejection with Total Turbine Bypass Failure									X	X						
Turbine Trip with Total Turbine Bypass Failure								X		X						
Control Rod Withdrawal Error During Refueling																
Control Rod Withdrawal Error During Startup													X			X
Control Rod Withdrawal Error During Power Operation																

Table 15.1-6

NSOA Automatic Instrument Trip/Event Matrix

	Scram – APRM High Neutron Flux	Scram – APRM High Simulated Thermal Power	Scram – RPV Low Water Level (L3)	Scram – RPV High Water Level (L8)	Scram – RPV High Dome Pressure	Scram – MSIV Closure	Scram – High Suppression Pool Temperature	Scram – TSV Closure (with TBP Inoperable)	Scram – TCV Fast Closure (with TBP Inoperable)	Scram – Insufficient Bypass Available	Scram – Low Condenser Vacuum	Scram – Feedwater not available	Scram – SRNM Period	Scram – High Drywell Pressure	Scram – Loss of two FW pumps (LOOP)	Rod Block – SRNM Period
Fuel Assembly Loading Error, Mislocated Bundle																
Fuel Assembly Loading Error, Misoriented Bundle																
Inadvertent SDC Function Operation	X															
Inadvertent Opening of a Safety/Relief Valve							X									
Inadvertent Opening of a DPV														X		
Stuck Open Safety/Relief Valve																
Liquid-Containing Tank Failure																

Table 15.1-6

NSOA Automatic Instrument Trip/Event Matrix

	Scram – APRM High Neutron Flux	Scram – APRM High Simulated Thermal Power	Scram – RPV Low Water Level (L3)	Scram – RPV High Water Level (L8)	Scram – RPV High Dome Pressure	Scram – MSIV Closure	Scram – High Suppression Pool Temperature	Scram – TSV Closure (with TBP Inoperable)	Scram – TCV Fast Closure (with TBP Inoperable)	Scram – Insufficient Bypass Available	Scram – Low Condenser Vacuum	Scram – Feedwater not available	Scram – SRNM Period	Scram – High Drywell Pressure	Scram – Loss of two FW pumps (LOOP)	Rod Block – SRNM Period
Fuel Handling Accident																
LOCA Inside Containment			X											X	X	
Main Steamline Break Outside Containment			X			X									X	
Control Rod Drop Accident																
Feedwater Line Break Outside Containment			X			X									X	
Failure of Small Line Outside Containment			X			X									X	
RWCU/SDC System Line Failure Outside Containment			X			X									X	
Spent Fuel Cask Drop Accident																

Table 15.1-6

NSOA Automatic Instrument Trip/Event Matrix

	Scram – APRM High Neutron Flux	Scram – APRM High Simulated Thermal Power	Scram – RPV Low Water Level (L3)	Scram – RPV High Water Level (L8)	Scram – RPV High Dome Pressure	Scram – MSIV Closure	Scram – High Suppression Pool Temperature	Scram – TSV Closure (with TBP Inoperable)	Scram – TCV Fast Closure (with TBP Inoperable)	Scram – Insufficient Bypass Available	Scram – Low Condenser Vacuum	Scram – Feedwater not available	Scram – SRNM Period	Scram – High Drywell Pressure	Scram – Loss of two FW pumps (LOOP)	Rod Block – SRNM Period
MSIV Closure Flux Scram (Overpressure Protection)	X															
Shutdown W/O Control Rods (SLCS capability)																
Shutdown from Outside Main Control Room																
Anticipated Transients Without Scram (ATWS)																
Station Blackout												X				
Safe Shutdown Fire																
Waste Gas System Leak or Failure																

Table 15.1-7

ESBWR NSOA Events

NSOA Event	Subsection Describing Event	Relevant Event Diagram
Loss of Feedwater Heating	15.2.1.1	15.1-3
Closure of One Turbine Control Valve	15.2.2.1	15.1-4
Generator Load Rejection with Turbine Bypass	15.2.2.2	15.1-5
Generator Load Rejection with a Single Failure in the Turbine Bypass System	15.2.2.3	15.1-6
Turbine Trip with Turbine Bypass	15.2.2.4	15.1-7
Turbine Trip with a Single Failure in the Turbine Bypass System	15.2.2.5	15.1-8
Closure of One Main Steam Isolation Valve	15.2.2.6	15.1-9
Closure of All Main Steam Isolation Valves	15.2.2.7	15.1-10
Loss of Condenser Vacuum	15.2.2.8	15.1-11
Loss of Shutdown Cooling Function of RWCU/SDC System	15.2.2.9	15.1-12
Inadvertent Isolation Condenser Initiation	15.2.4.1	15.1-13
Runout of One Feedwater Pump	15.2.4.2	15.1-14
Opening of One Turbine Control or Bypass Valve	15.2.5.1	15.1-15
Loss of Unit Auxiliary Transformer	15.2.5.2	15.1-16
Loss of Grid Connection	15.2.5.2	15.1-17
Loss of All Feedwater Flow	15.2.5.3	15.1-18
Loss of Feedwater Heating With Failure of Selected Control Rod Run-In	15.3.1	15.1-19
Feedwater Controller Failure – Maximum Demand	15.3.2	15.1-20
Pressure Regulator Failure Opening of All Turbine Control and Bypass Valves	15.3.3	15.1-21
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves	15.3.4	15.1-22
Generator Load Rejection with Total Turbine Bypass Failure	15.3.5	15.1-23
Turbine Trip with Total Turbine Bypass Failure	15.3.6	15.1-24
Control Rod Withdrawal Error During Refueling	15.3.7	15.1-25

Table 15.1-7

ESBWR NSOA Events

NSOA Event	Subsection Describing Event	Relevant Event Diagram
Control Rod Withdrawal Error During Startup	15.3.8	15.1-26
Control Rod Withdrawal Error During Power Operation	15.3.9	15.1-27
Fuel Assembly Loading Error, Mislocated Bundle	15.3.10	15.1-28
Fuel Assembly Loading Error, Misoriented Bundle	15.3.11	15.1-29
Inadvertent SDC Function Operation	15.3.12	15.1-30
Inadvertent Opening of a Safety/Relief Valve	15.3.13	15.1-31
Inadvertent Opening of a Depressurization Valve	15.3.14	15.1-32
Stuck Open Safety/Relief Valve	15.3.15	15.1-33
Liquid-Containing Tank Failure	15.3.16	15.1-34
Fuel Handling Accident	15.4.1	15.1-35
LOCA Inside Containment	15.4.2, 15.4.3, 15.4.4	15.1-36
Main Steamline Break Outside Containment	15.4.5	15.1-37
Control Rod Drop Accident	15.4.6	15.1-38
Feedwater Line Break Outside Containment	15.4.7	15.1-39
Failure of Small Line Carrying Primary Coolant Outside Containment	15.4.8	15.1-40
RWCU/SDC System Line Failure Outside Containment	15.4.9	15.1-41
Spent Fuel Cask Drop Accident	15.4.10	15.1-42
MSIV Closure with Flux Scram (Overpressure Protection)	15.5.1	15.1-43
Shutdown Without Control Rods (i.e., SLCS shutdown capability)	15.5.2	15.1-44
Shutdown from Outside Main Control Room	15.5.3	15.1-45
Anticipated Transients Without Scram	15.5.4	15.1-46
Station Blackout	15.5.5	15.1-47
Safe Shutdown Fire	15.5.6	15.1-48
Waste Gas System Leak or Failure	15.5.7	15.1-49

Figure 15.1-1. Event Diagram Format

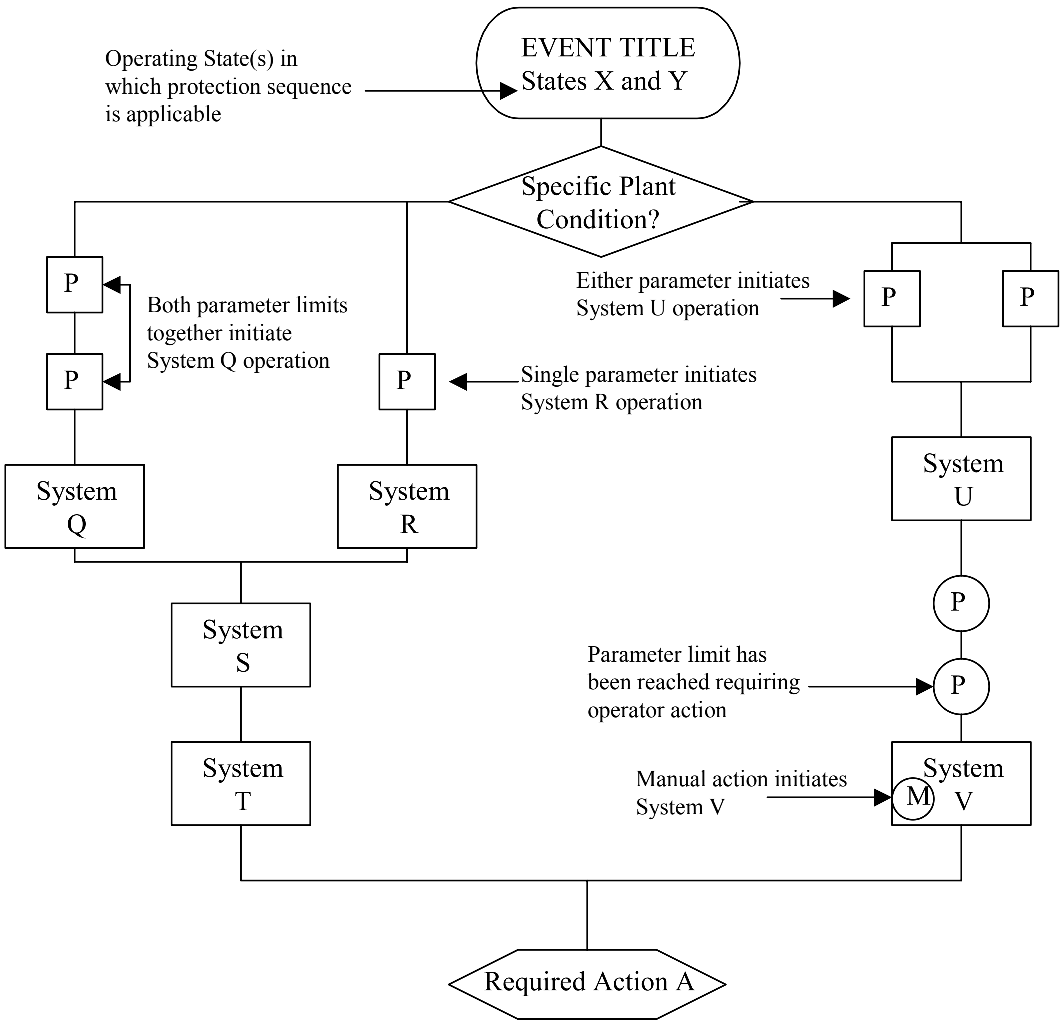
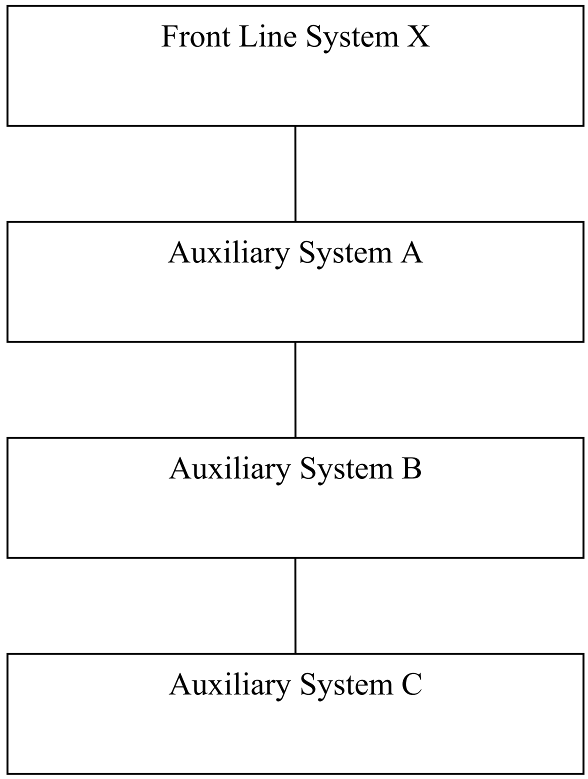


Figure 15.1-2. Auxiliary System Diagram Format



Auxiliary Systems A, B, and C are required for operation of Front Line System X.

No chronological order of actions is implied.

Figure 15.1-3. Event Diagram – Loss of Feedwater Heating

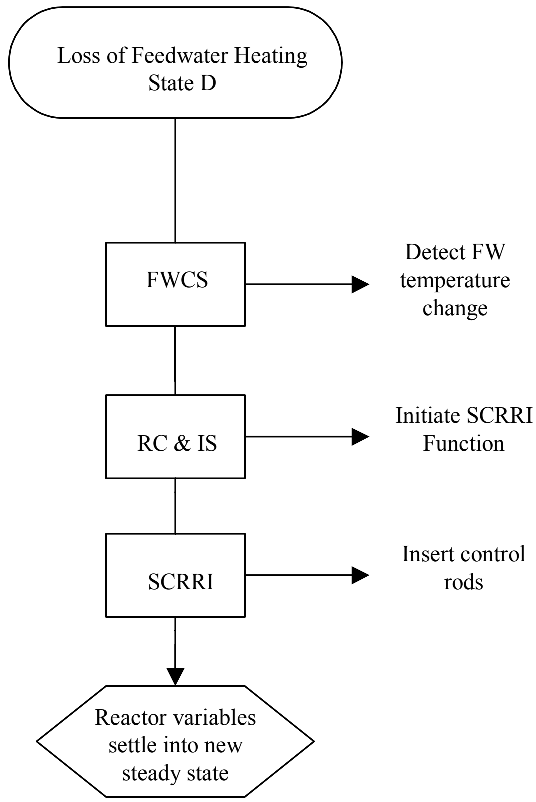


Figure 15.1-4. Event Diagram – Closure of One Turbine Control Valve

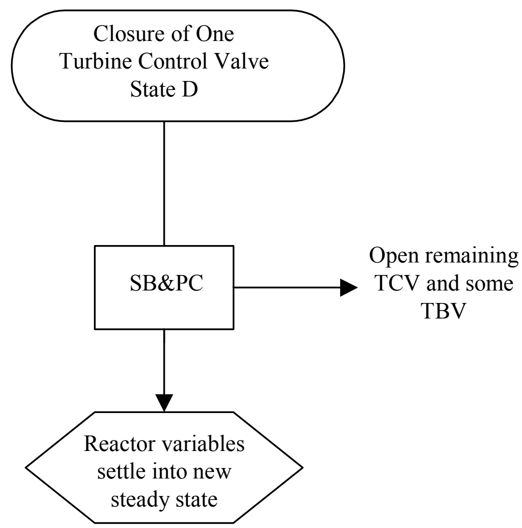


Figure 15.1-5. Event Diagram – Generator Load Rejection with Turbine Bypass

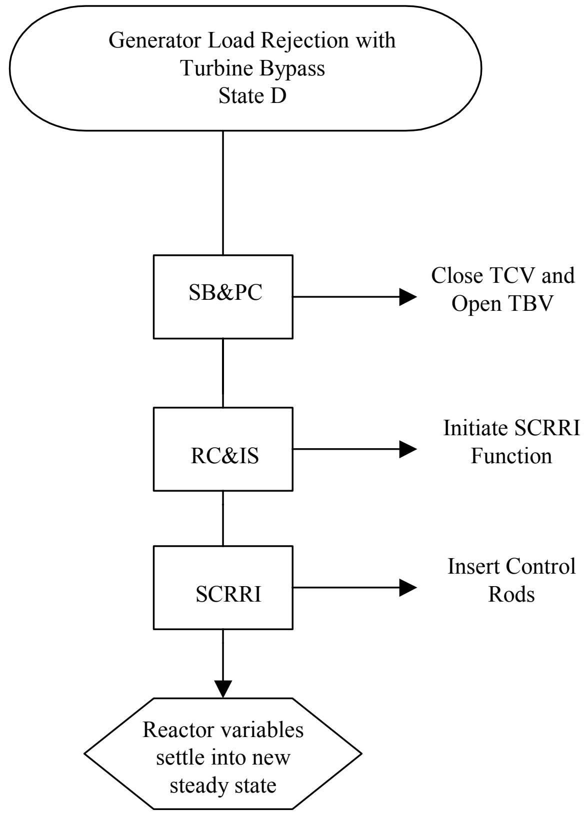


Figure 15.1-6. Event Diagram – Generator Load Rejection with a Single Failure in the Turbine Bypass System

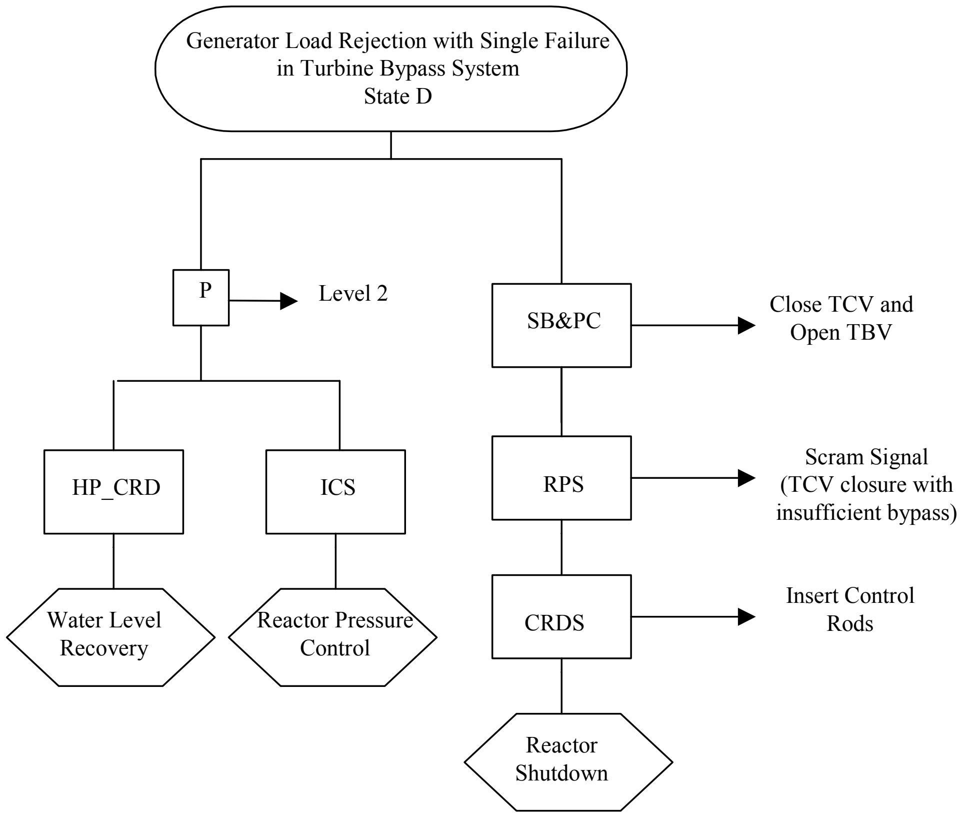


Figure 15.1-7. Event Diagram – Turbine Trip with Turbine Bypass

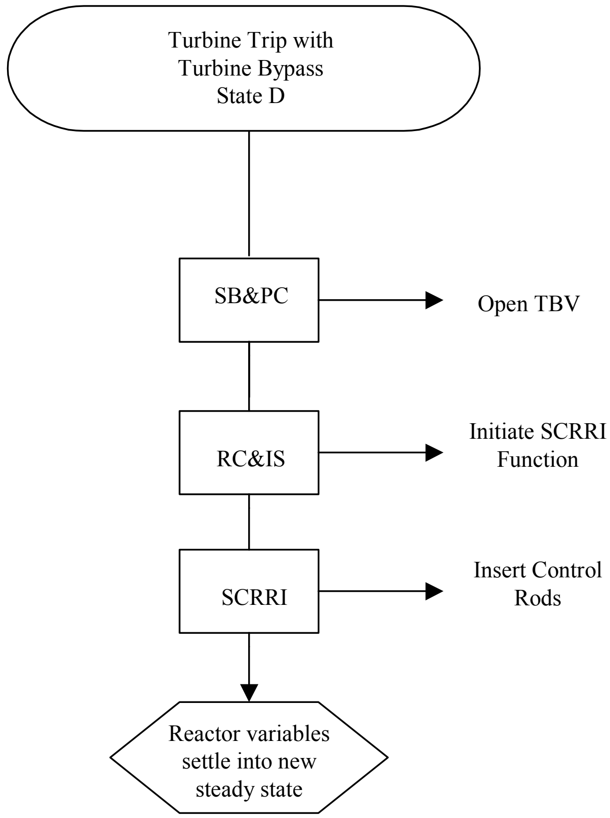


Figure 15.1-8. Event Diagram – Turbine Trip with a Single Failure in the Turbine Bypass System

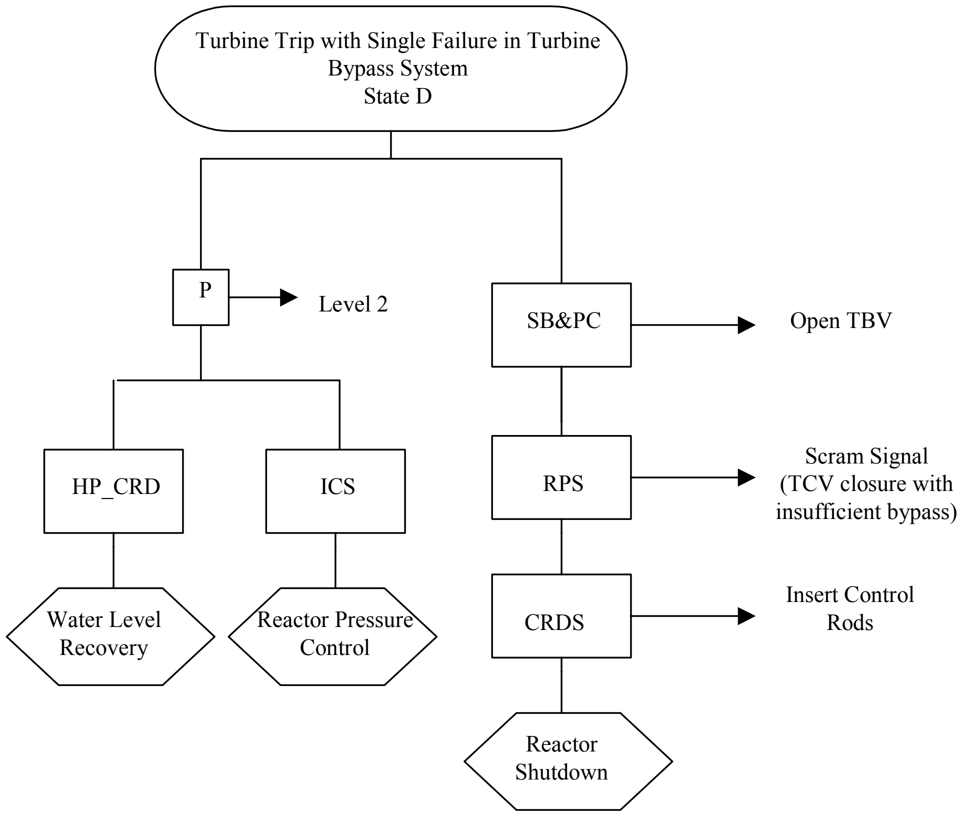


Figure 15.1-9. Event Diagram – Closure of One Main Steam Isolation Valve

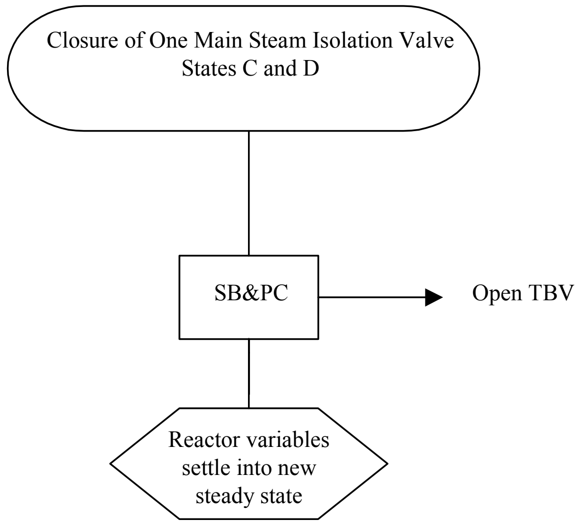


Figure 15.1-10. Event Diagram – Closure of All Main Steam Isolation Valves

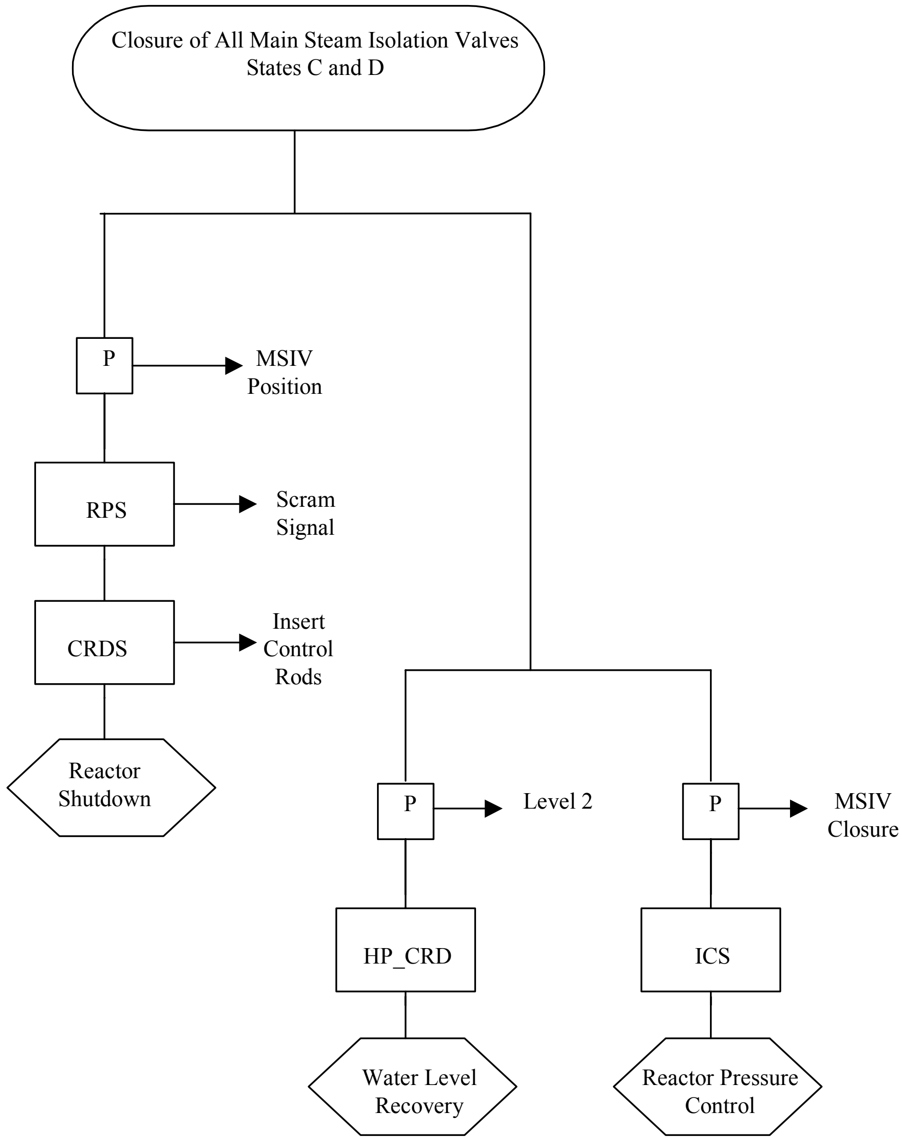


Figure 15.1-11. Event Diagram – Loss of Condenser Vacuum

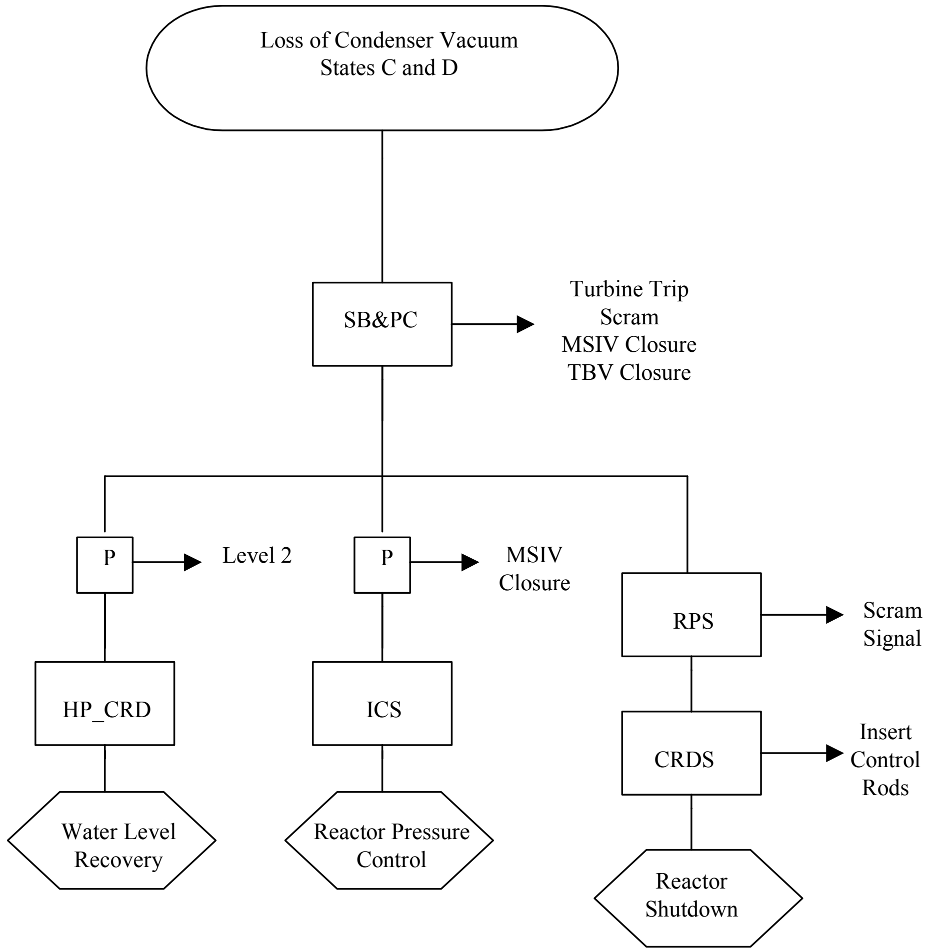


Figure 15.1-12. Event Diagram – Loss of Shutdown Cooling Function of RWCU/SDC System

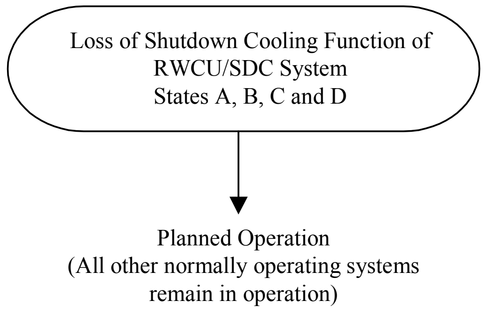


Figure 15.1-13. Event Diagram – Inadvertent Isolation Condenser Operation

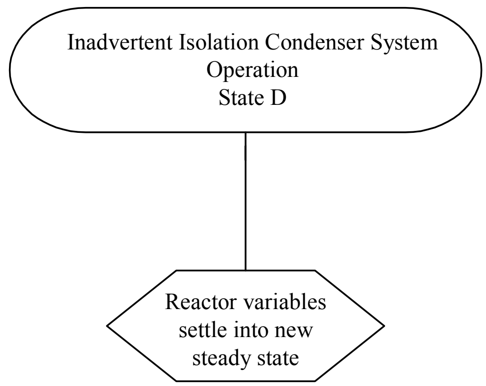


Figure 15.1-14. Event Diagram – Runout of One Feedwater Pump

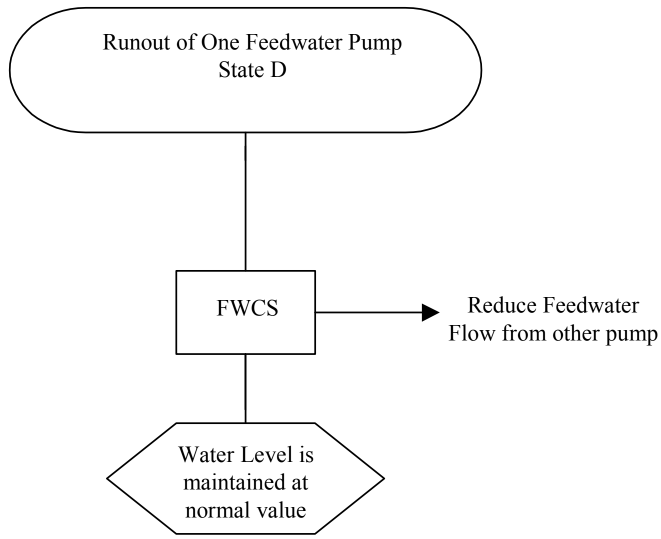


Figure 15.1-15. Event Diagram – Opening of One Turbine Control or Bypass Valve

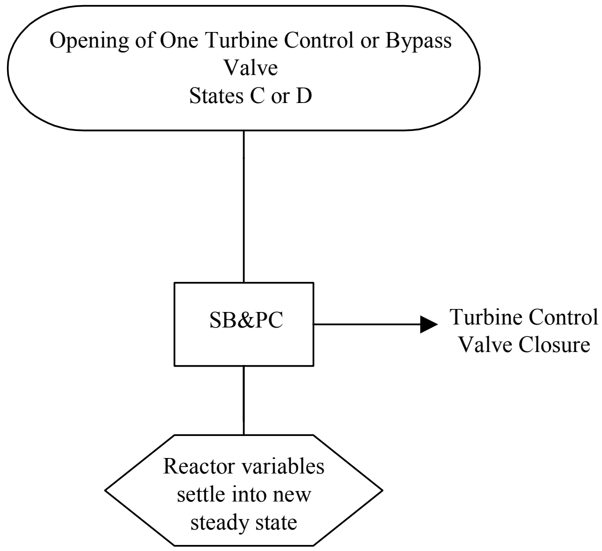


Figure 15.1-16. Event Diagram – Loss of Unit Auxiliary Transformer

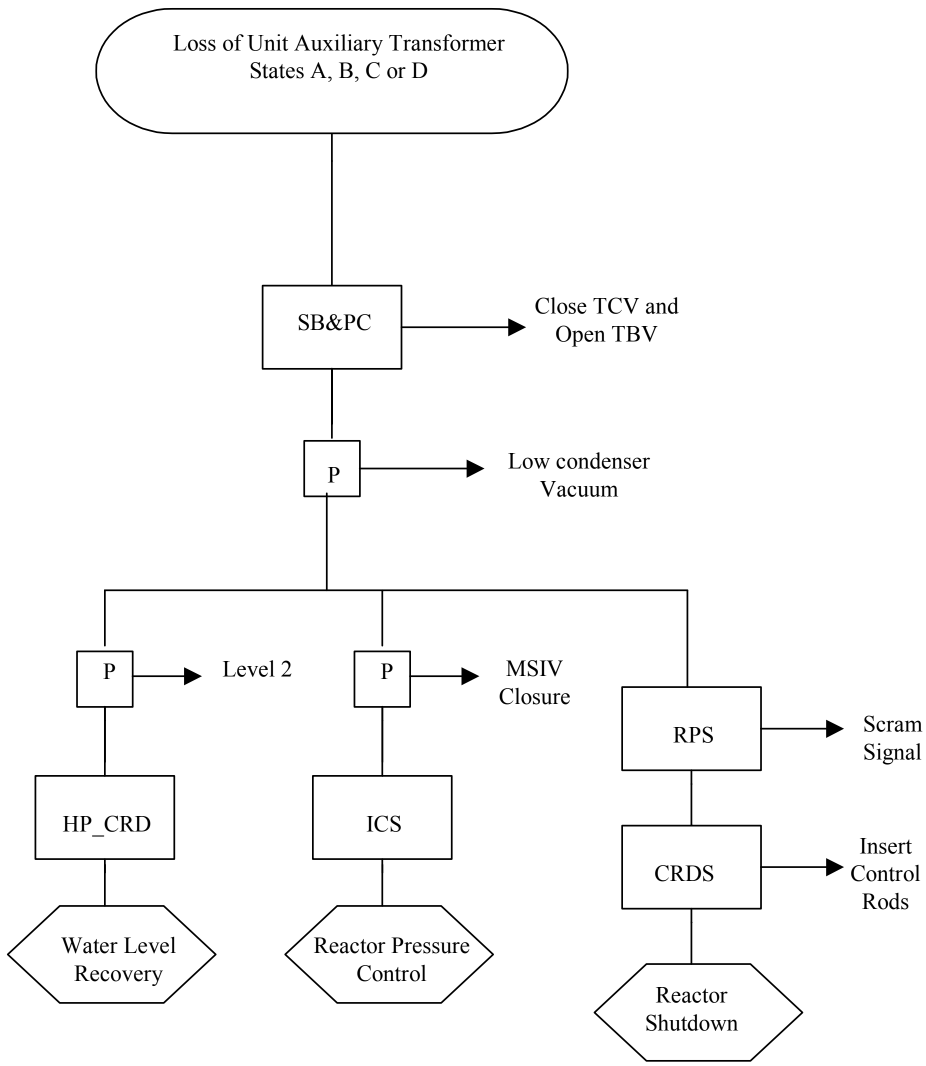


Figure 15.1-17. Event Diagram – Loss of Grid Connection

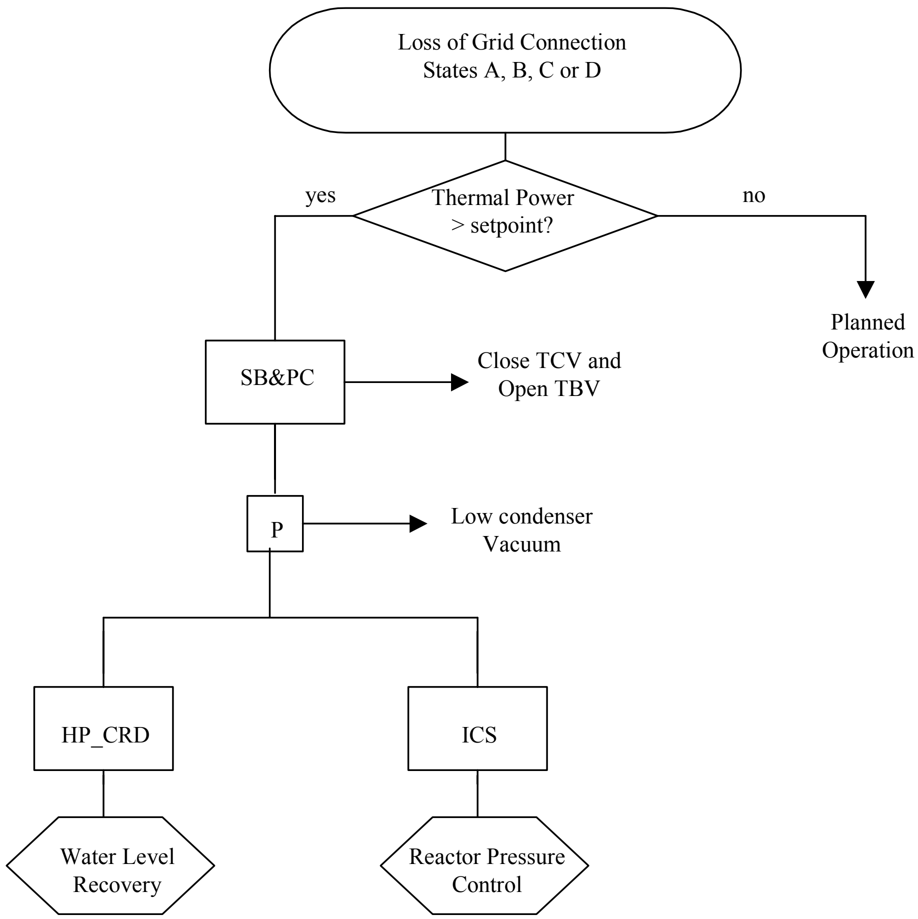


Figure 15.1-18. Event Diagram – Loss of All Feedwater Flow

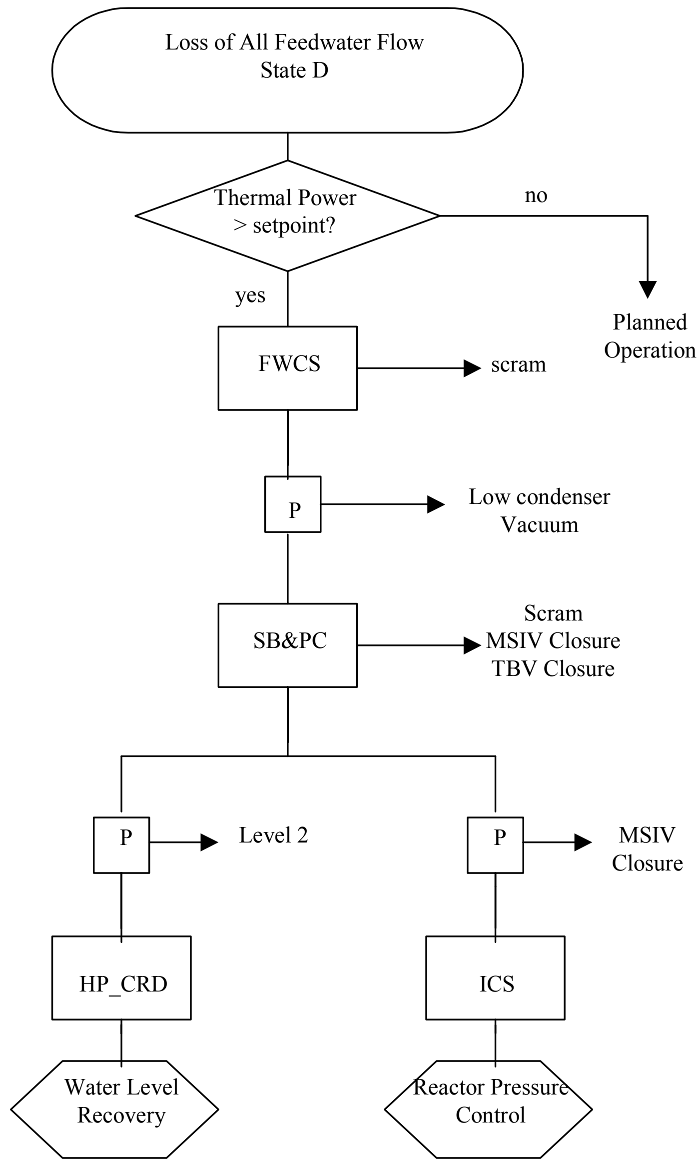


Figure 15.1-19. Event Diagram – Loss of Feedwater Heating With Failure of Selected Control Rod Run-In

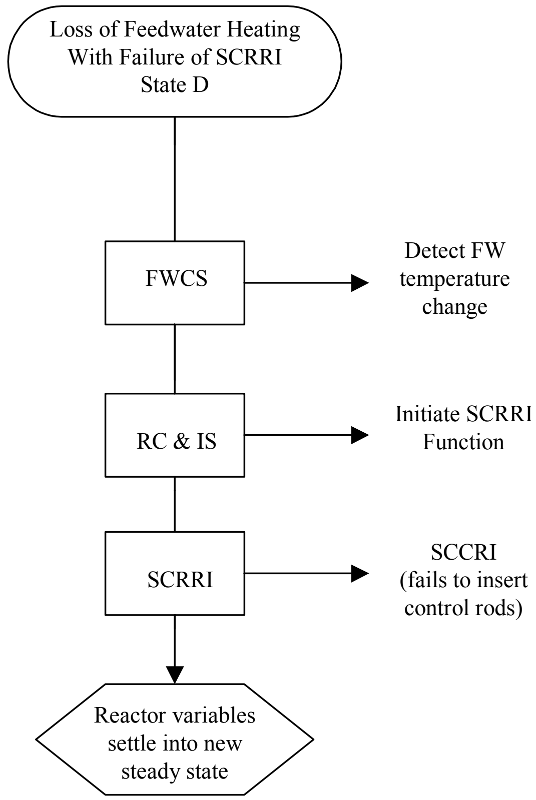


Figure 15.1-20. Event Diagram – Feedwater Controller Failure – Maximum Demand

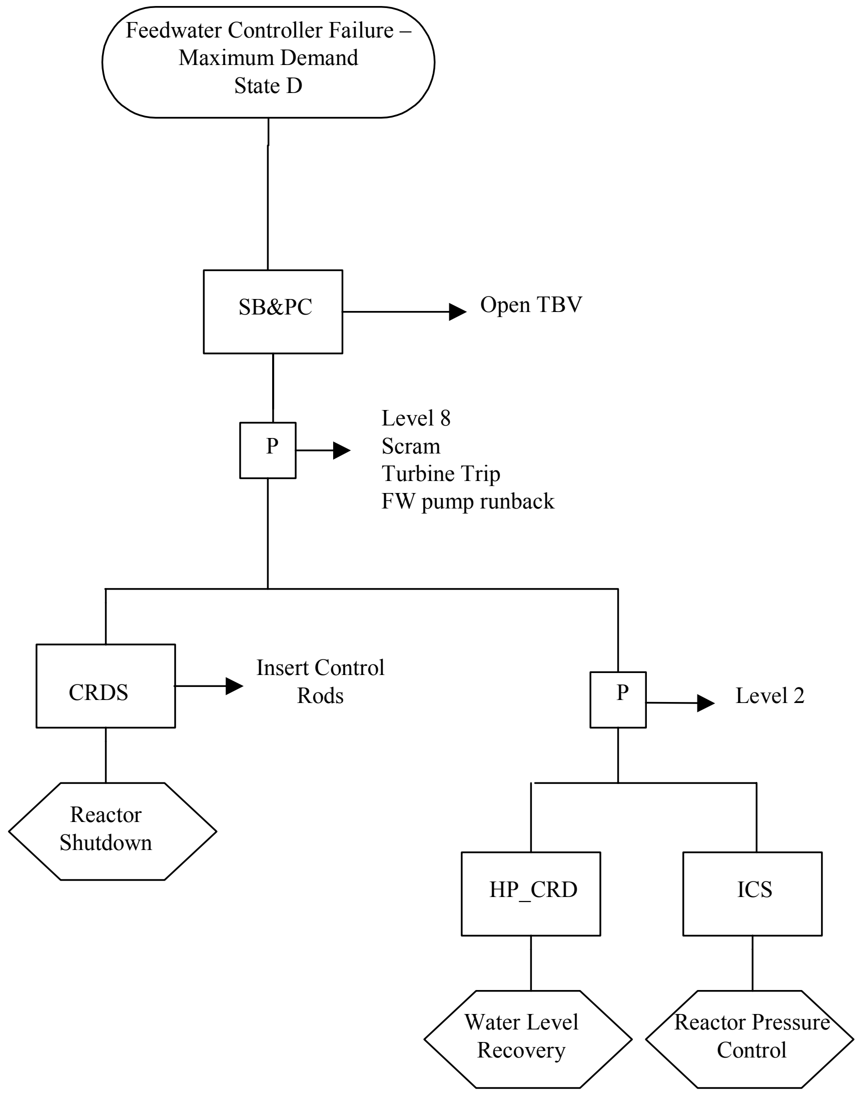


Figure 15.1-21. Event Diagram – Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

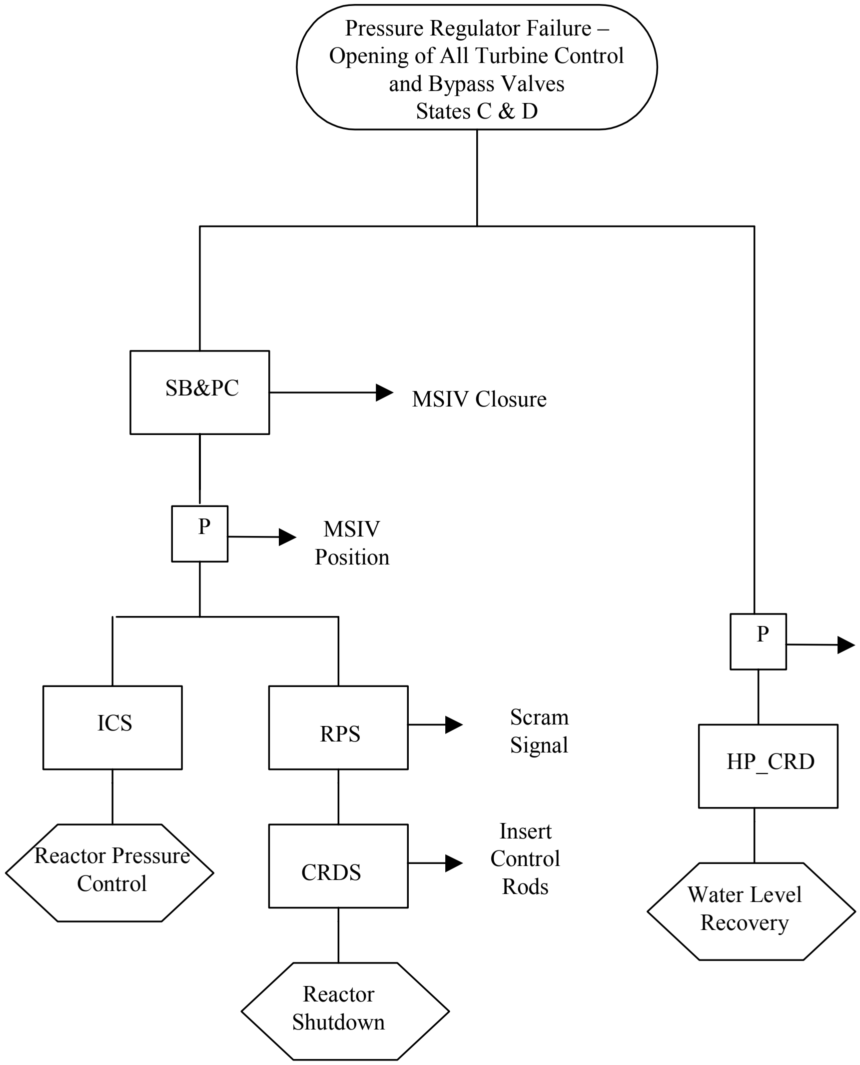


Figure 15.1-22. Event Diagram – Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

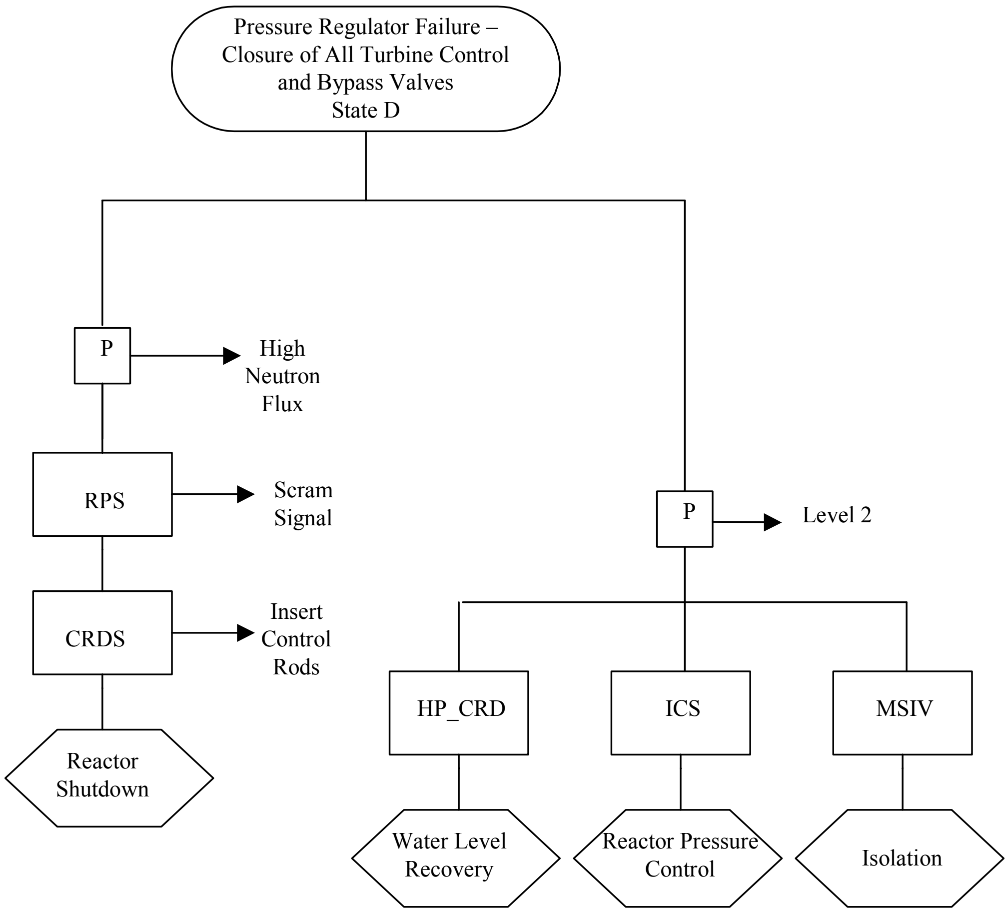


Figure 15.1-23. Event Diagram – Generator Load Rejection with Total Bypass Failure

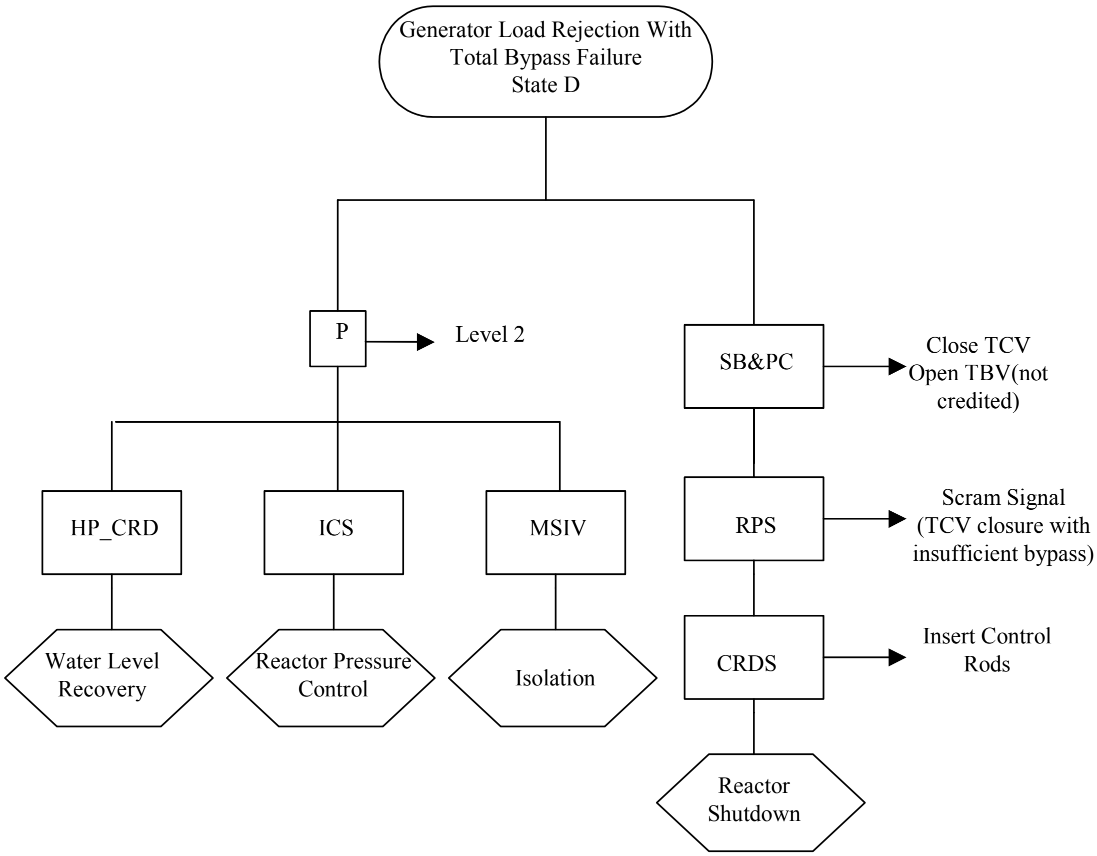


Figure 15.1-24. Event Diagram – Turbine Trip with Total Bypass Failure

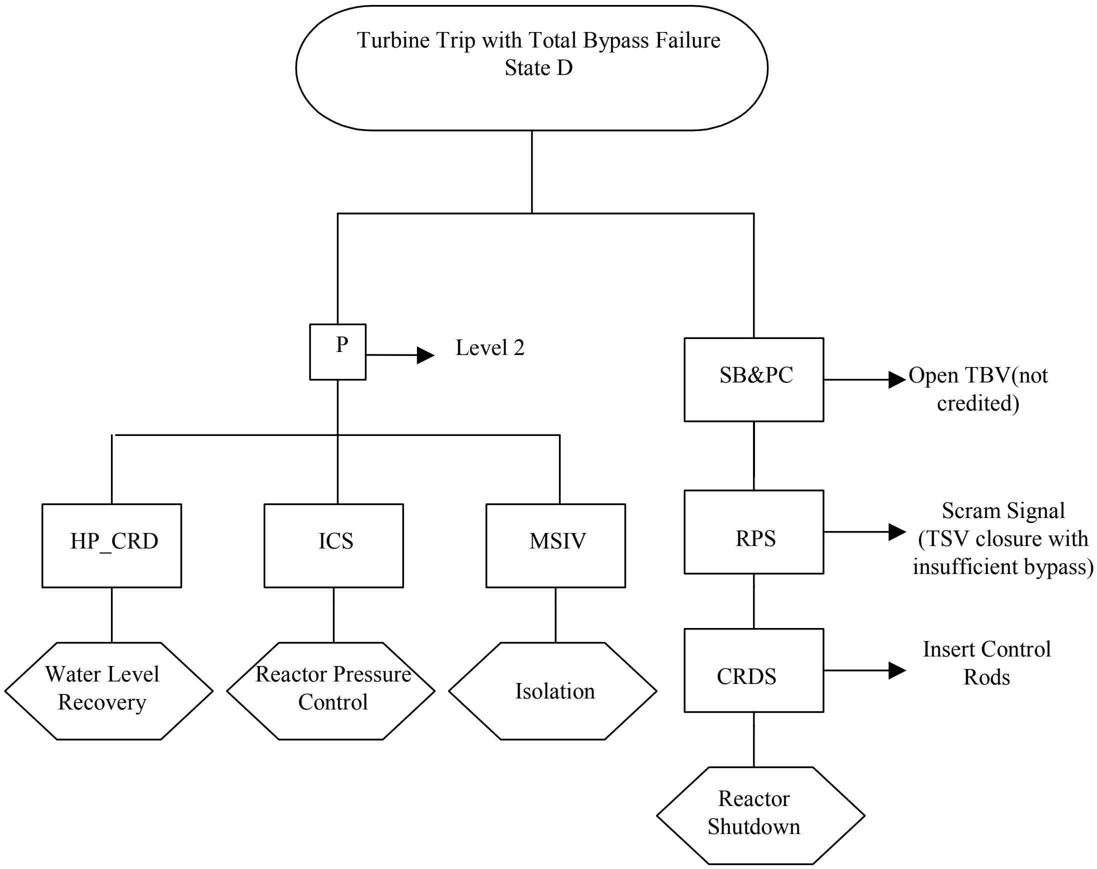


Figure 15.1-25. Event Diagram – Control Rod Withdrawal Error During Refueling

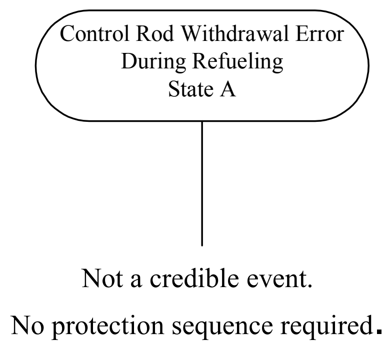


Figure 15.1-26. Event Diagram – Control Rod Withdrawal Error During Startup

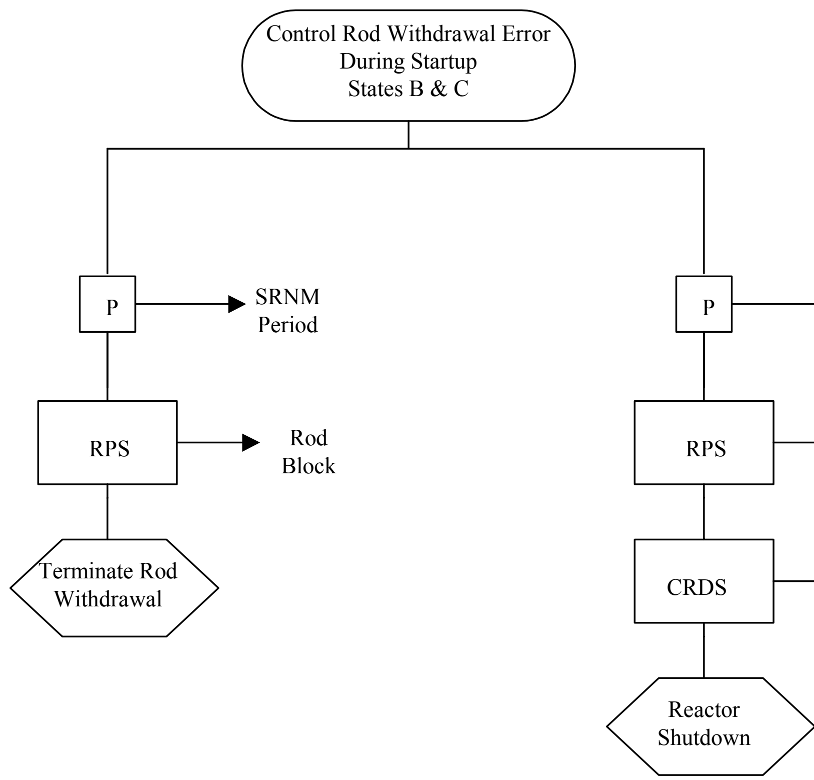


Figure 15.1-27. Event Diagram – Control Rod Withdrawal Error During Power Operation

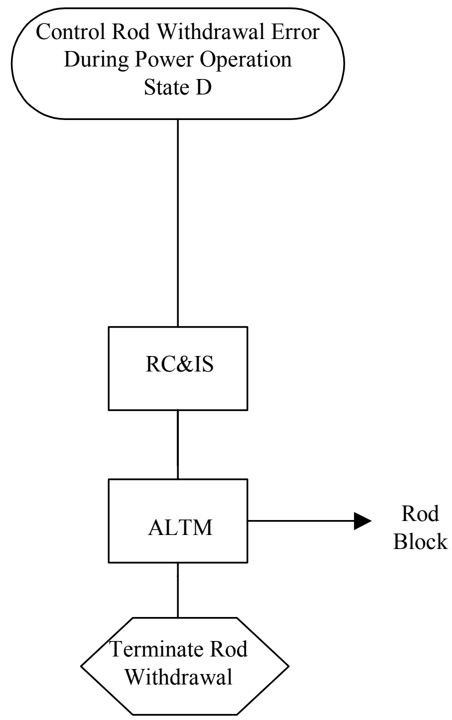


Figure 15.1-28. Event Diagram – Fuel Loading Error – Mislocated Bundle

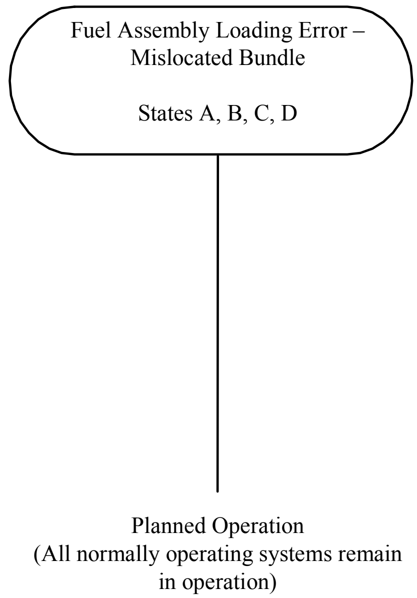


Figure 15.1-29. Event Diagram – Fuel Loading Error – Misoriented Bundle

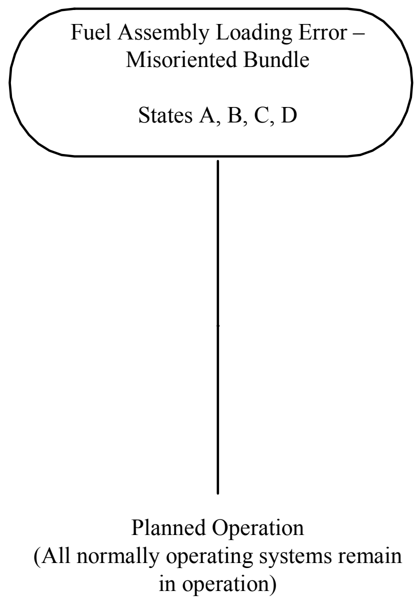


Figure 15.1-30. Event Diagram – Inadvertent SDC Function Operation

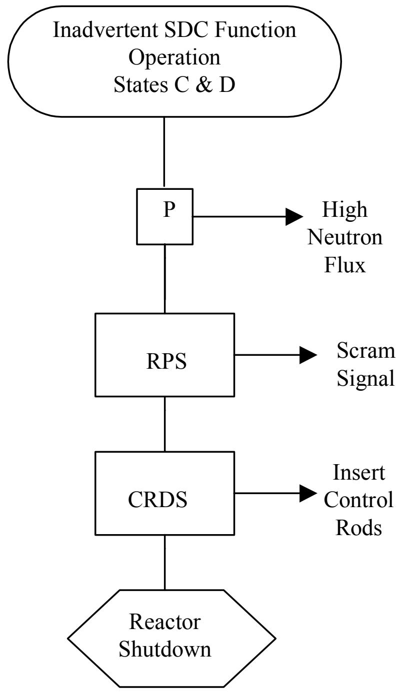


Figure 15.1-31. Event Diagram – Inadvertent Opening of a Safety/Relief Valve

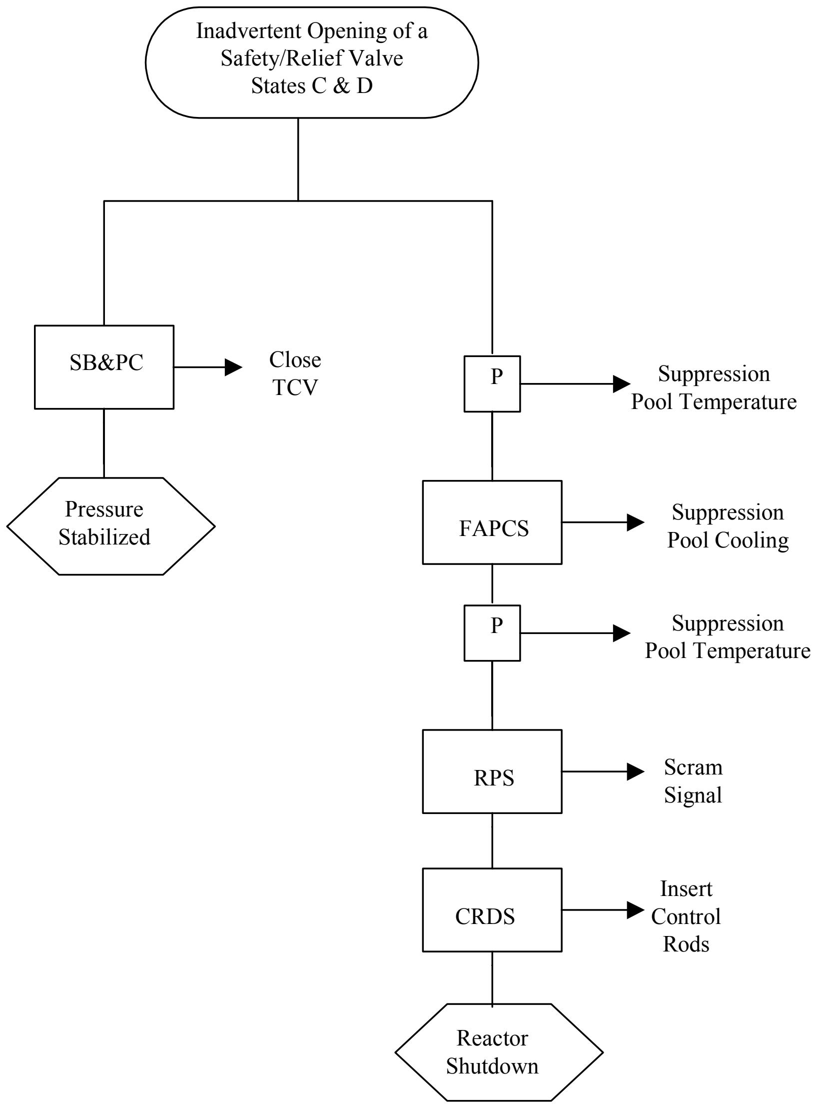


Figure 15.1-32. Event Diagram – Inadvertent Opening of a Depressurization Valve

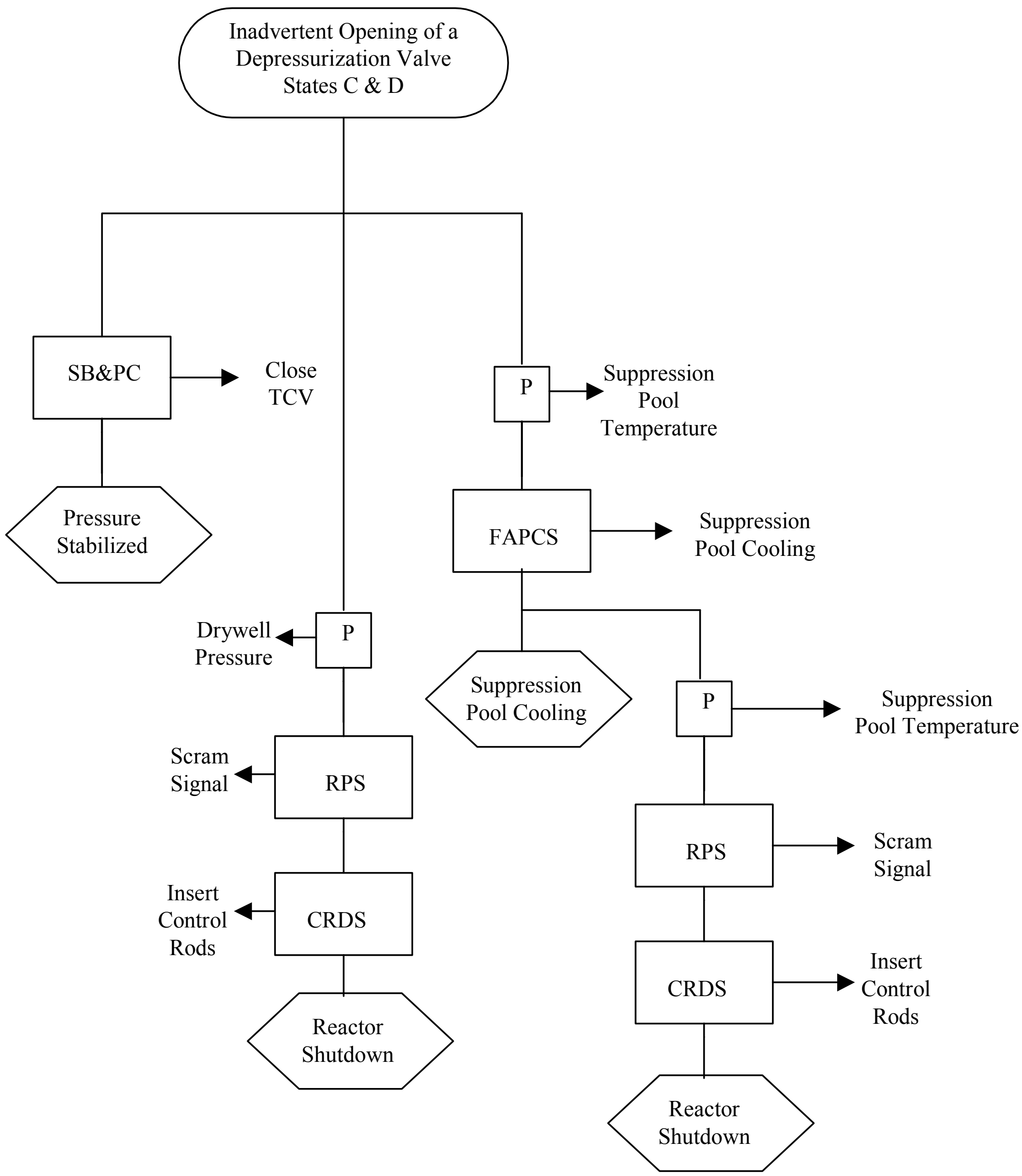


Figure 15.1-33. Event Diagram – Stuck Open Safety/Relief Valve

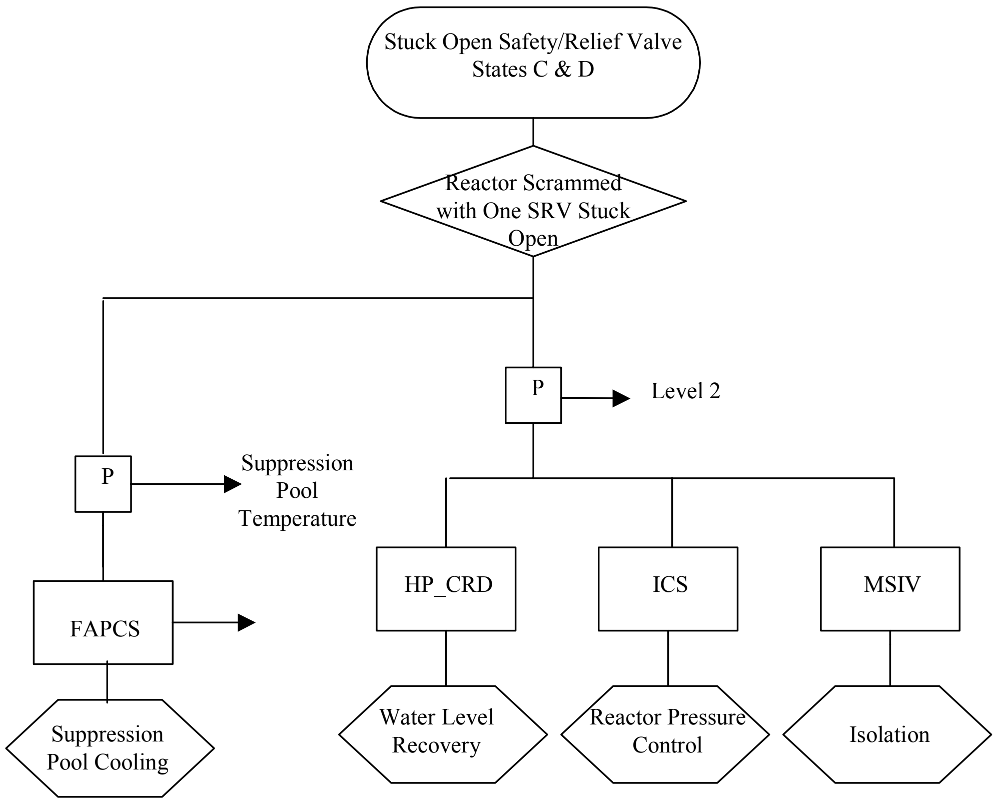


Figure 15.1-34. Event Diagram – Liquid Containing Tank Failure

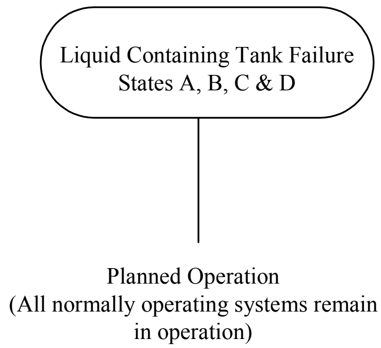


Figure 15.1-35. Event Diagram – Fuel Handling Accident

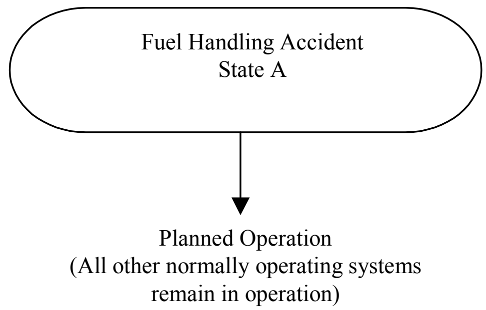


Figure 15.1-36a. Event Diagram – Loss-of-Coolant Accident Inside Containment

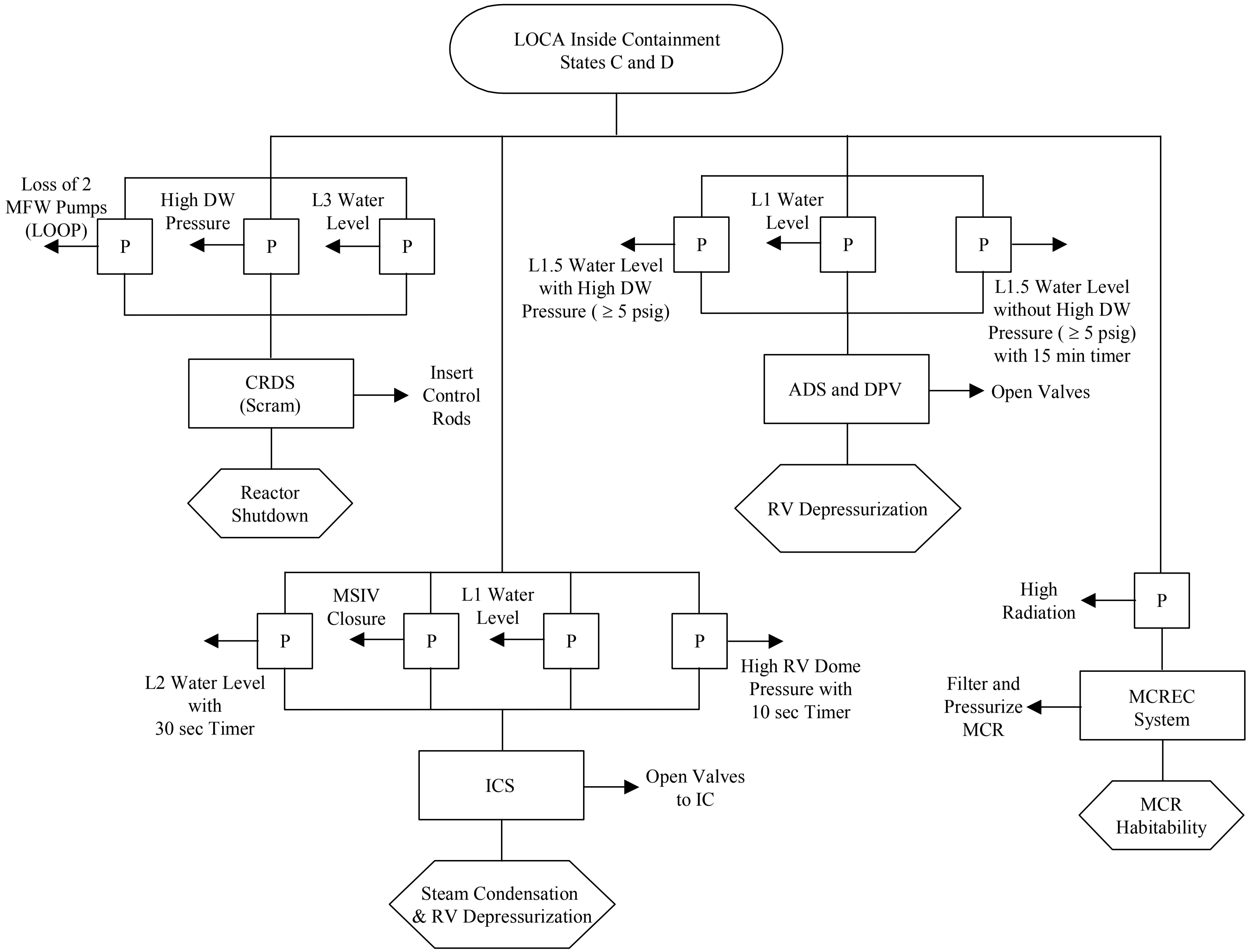


Figure 15.1-36b. Event Diagram – Loss-of-Coolant Accident Inside Containment

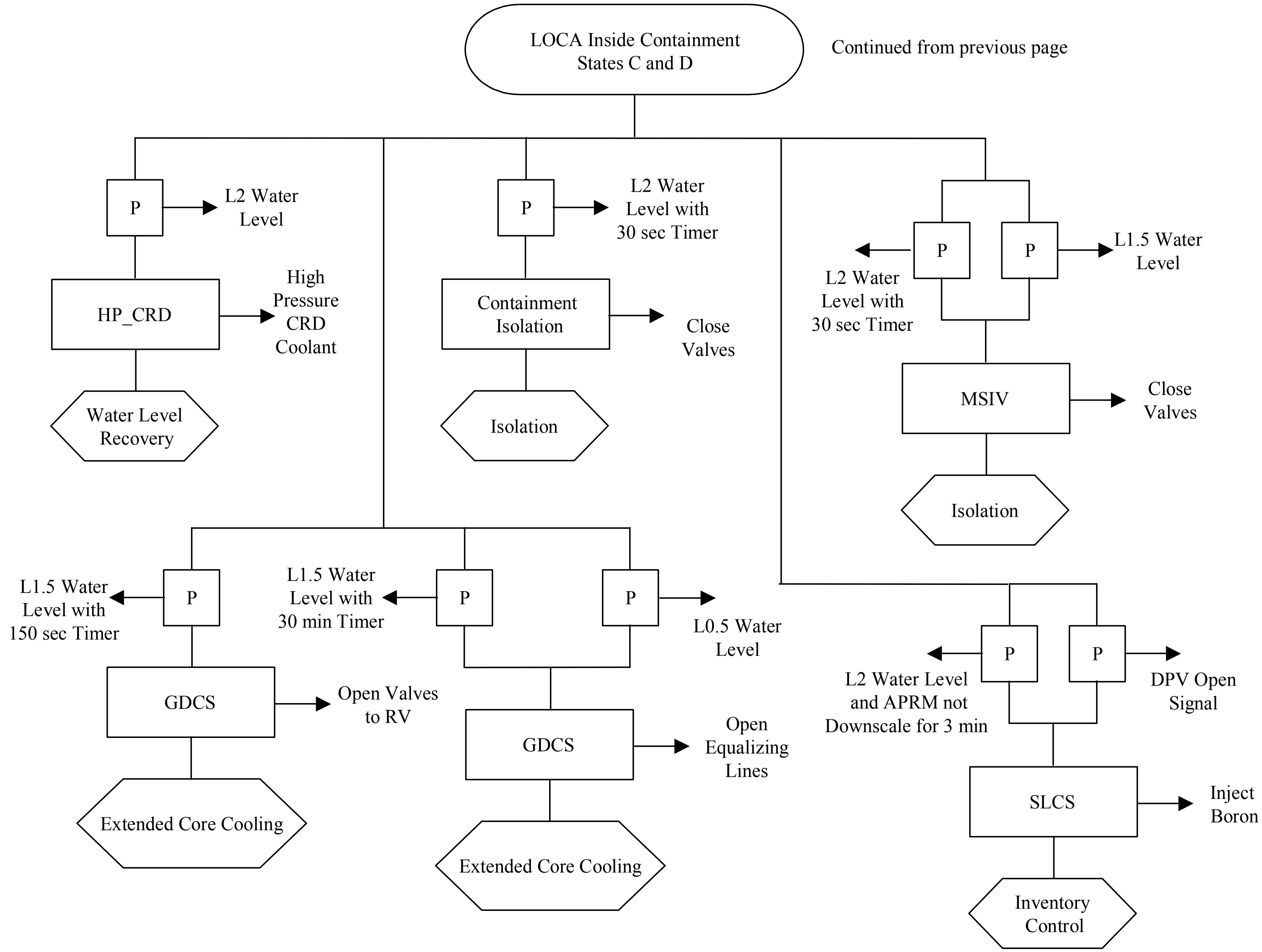


Figure 15.1-37a. Event Diagram – Main Steamline Break Outside Containment

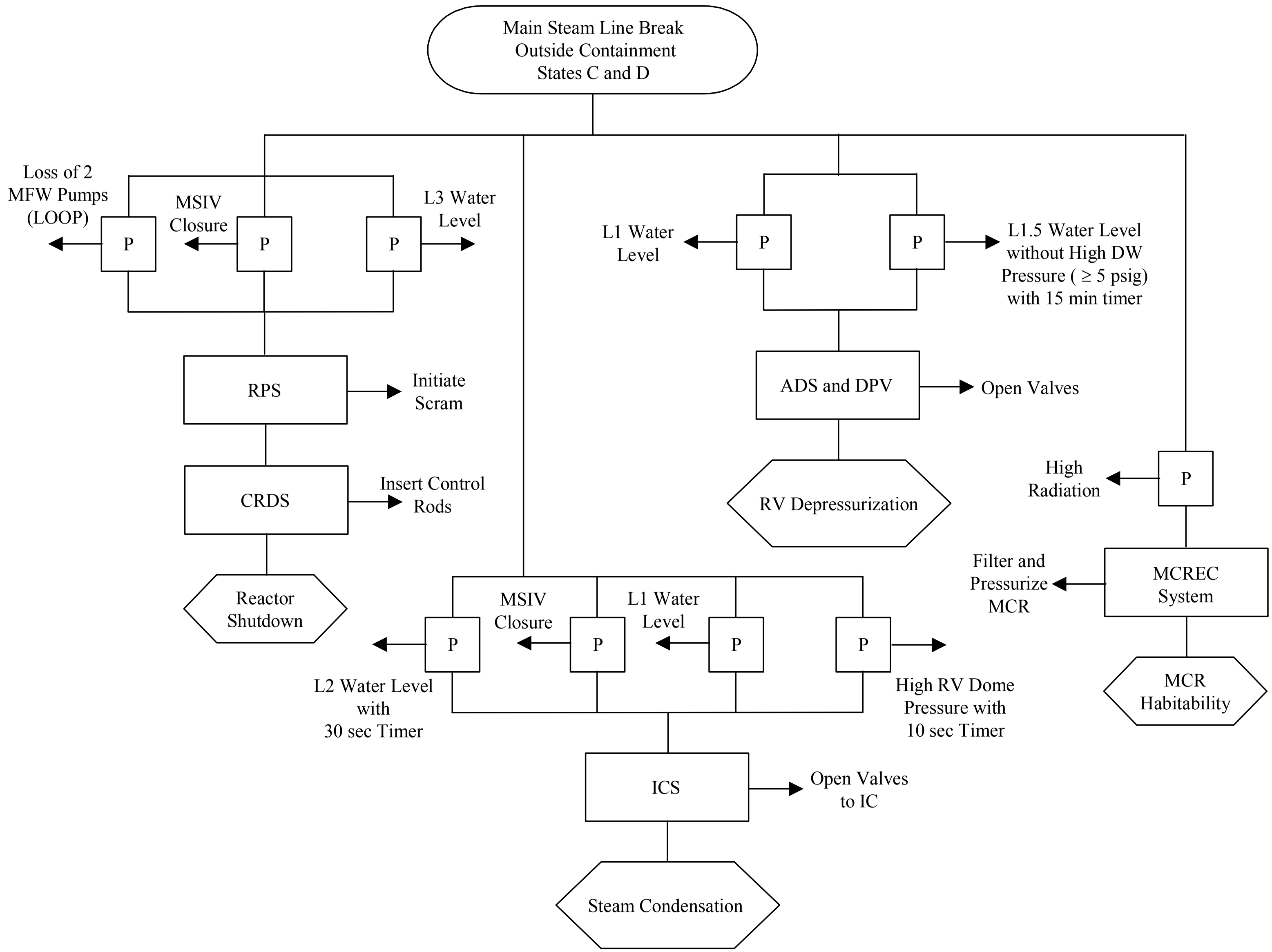


Figure 15.1-37b. Event Diagram – Main Steamline Break Outside Containment

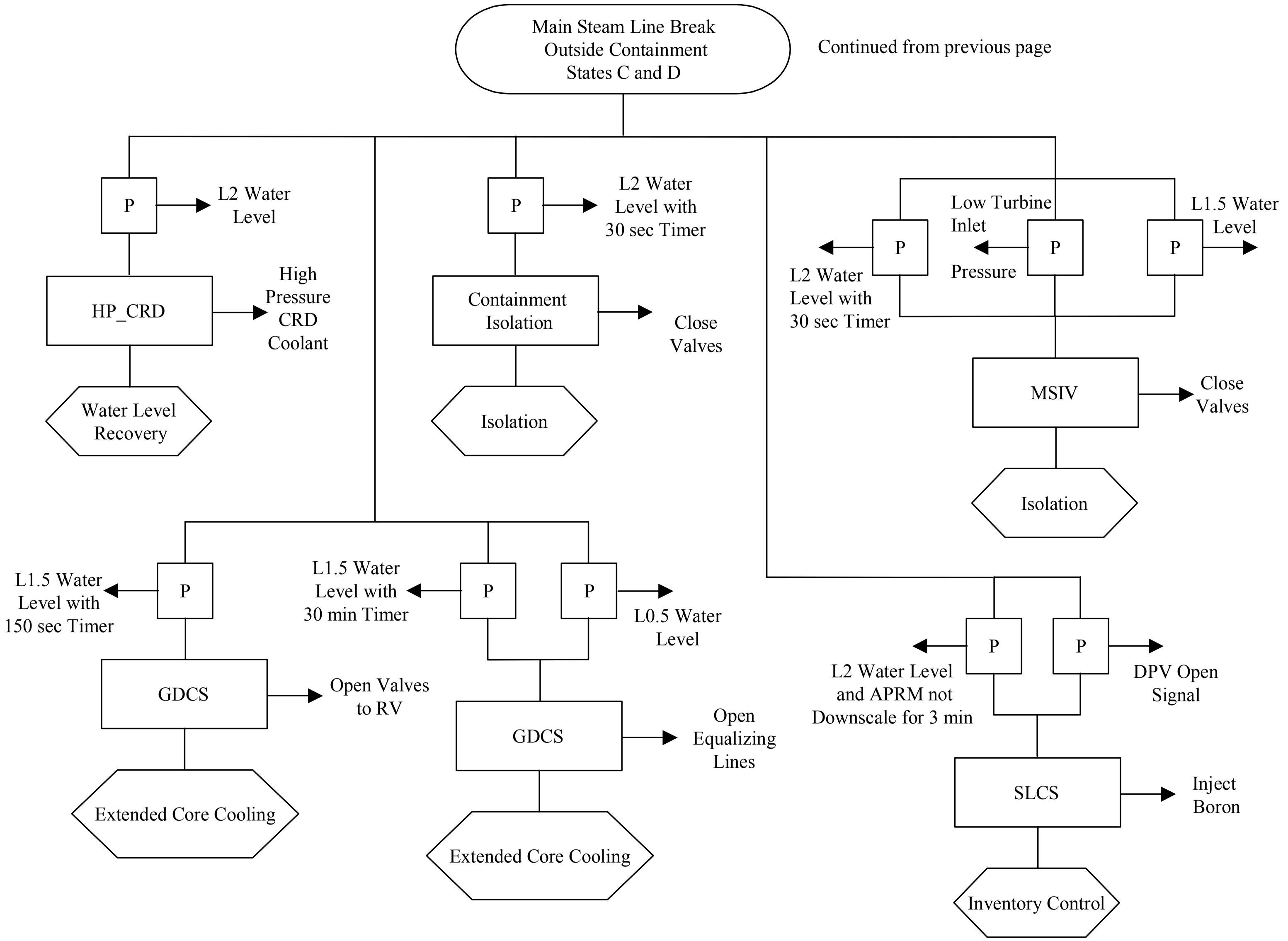


Figure 15.1-38. Event Diagram – Control Rod Drop Accident

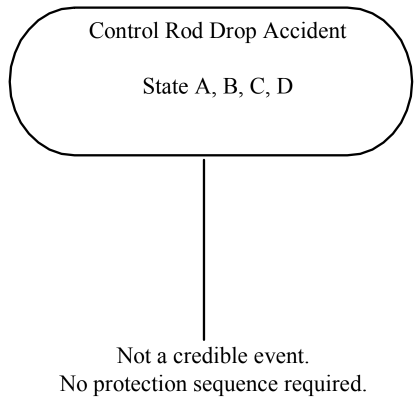


Figure 15.1-39a. Event Diagram – Feedwater Line Break Outside Containment

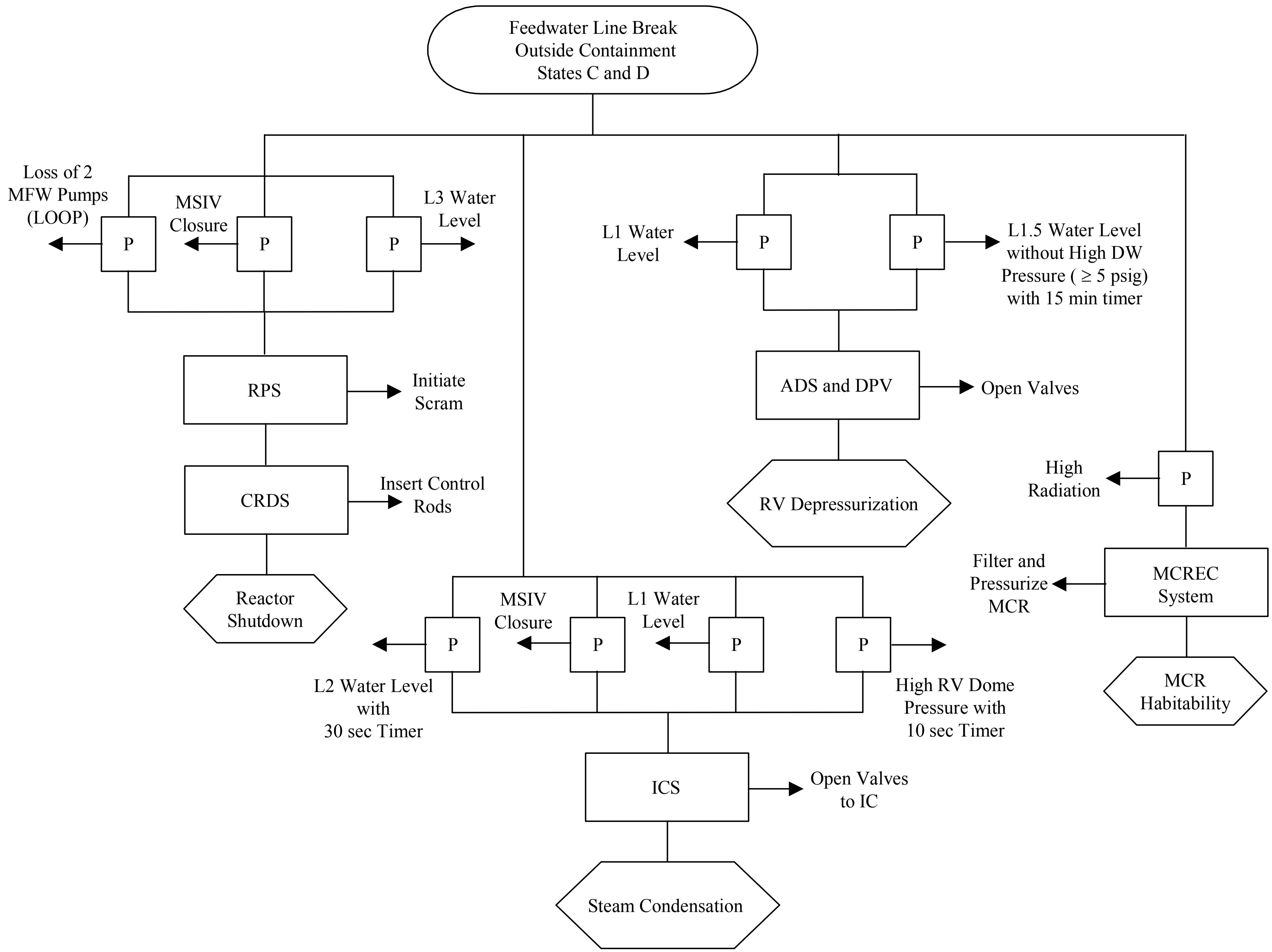


Figure 15.1-39b. Event Diagram – Feedwater Line Break Outside Containment

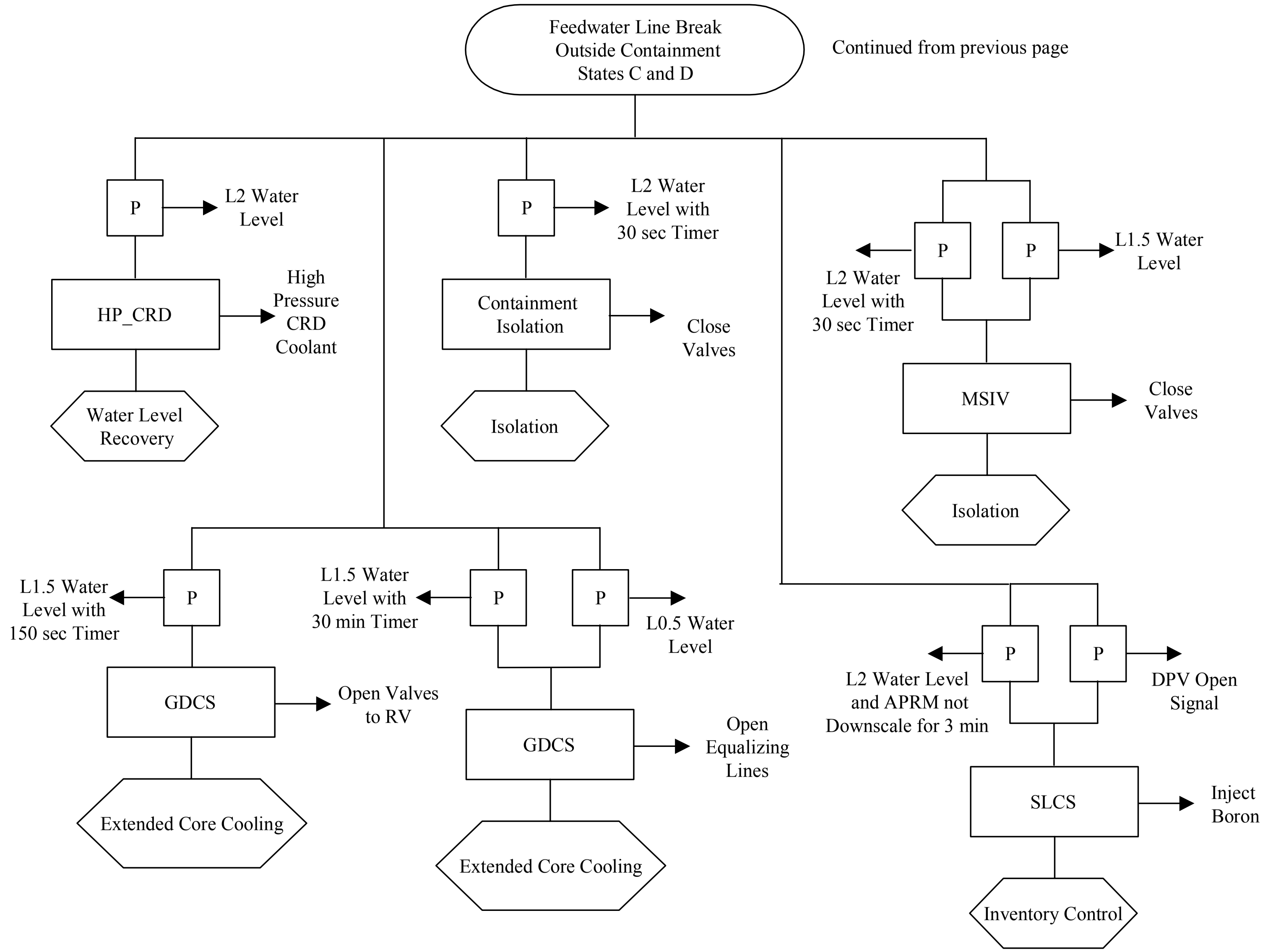


Figure 15.1-40a. Event Diagram – Failure of Small Line Carrying Primary Coolant Outside Containment

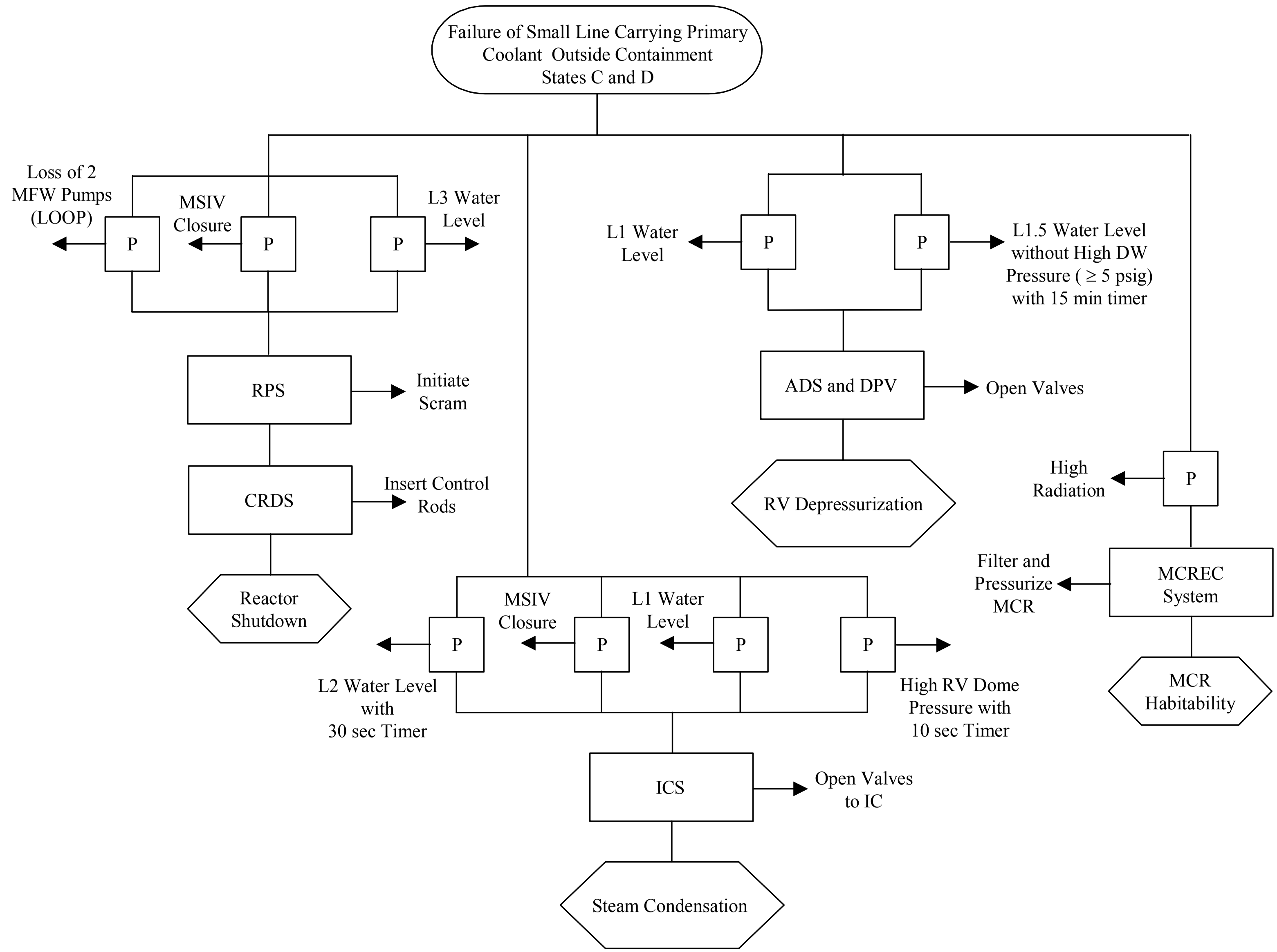


Figure 15.1-40b. Event Diagram – Failure of Small Line Carrying Primary Coolant Outside Containment

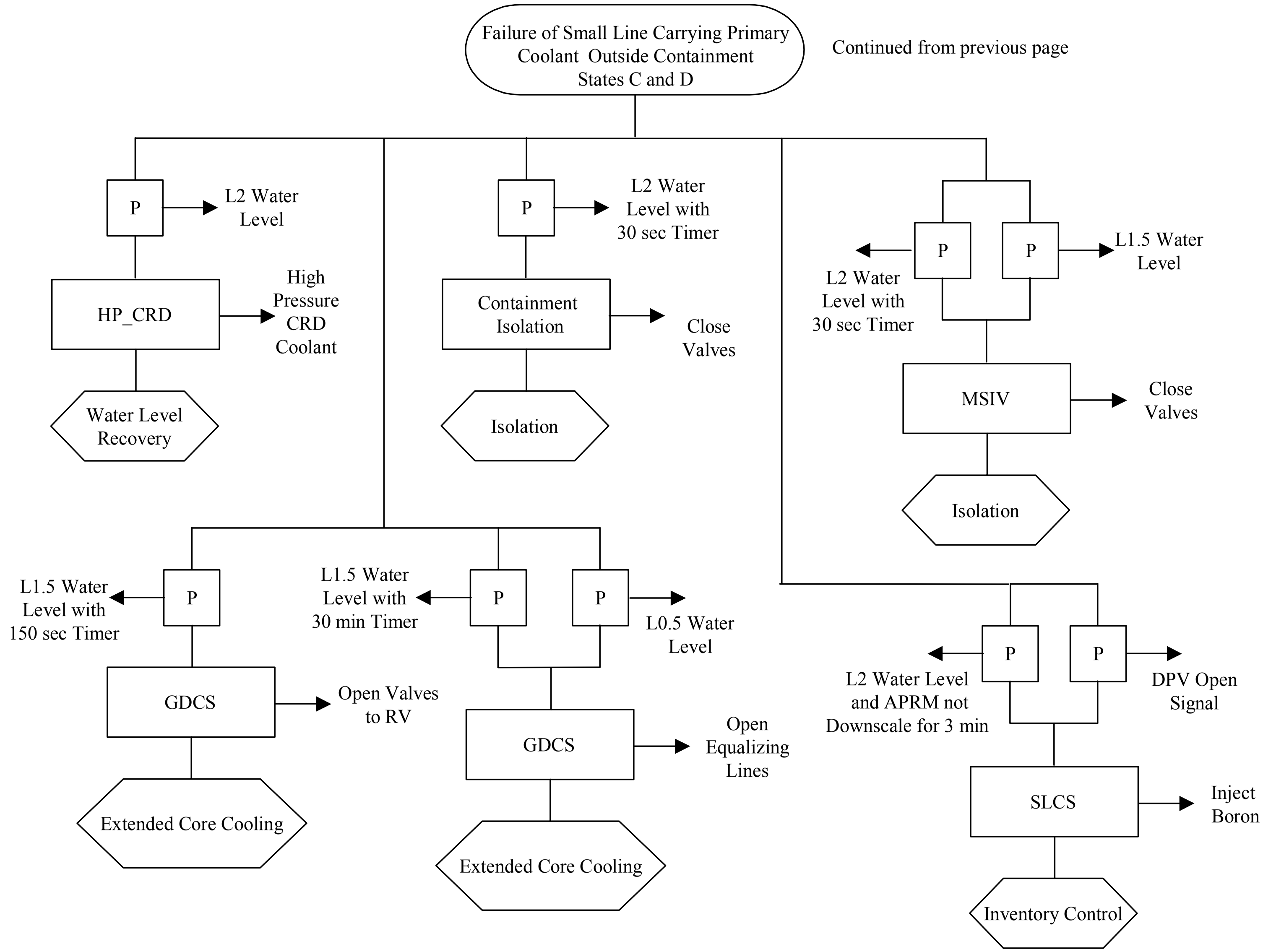


Figure 15.1-41a. Event Diagram – RWCU/SDC System Line Failure Outside Containment

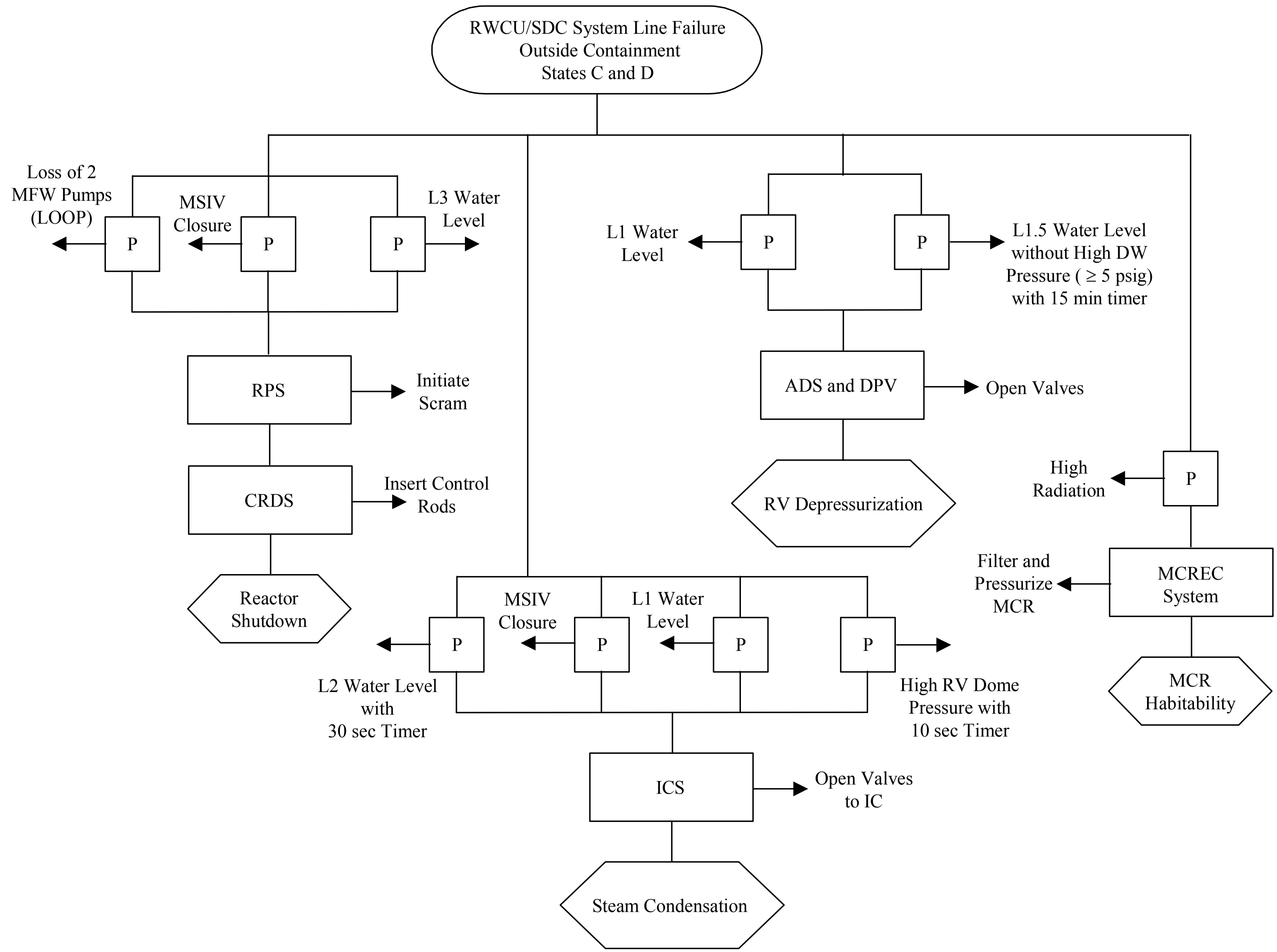


Figure 15.1-41b. Event Diagram – RWCU/SDC System Line Failure Outside Containment

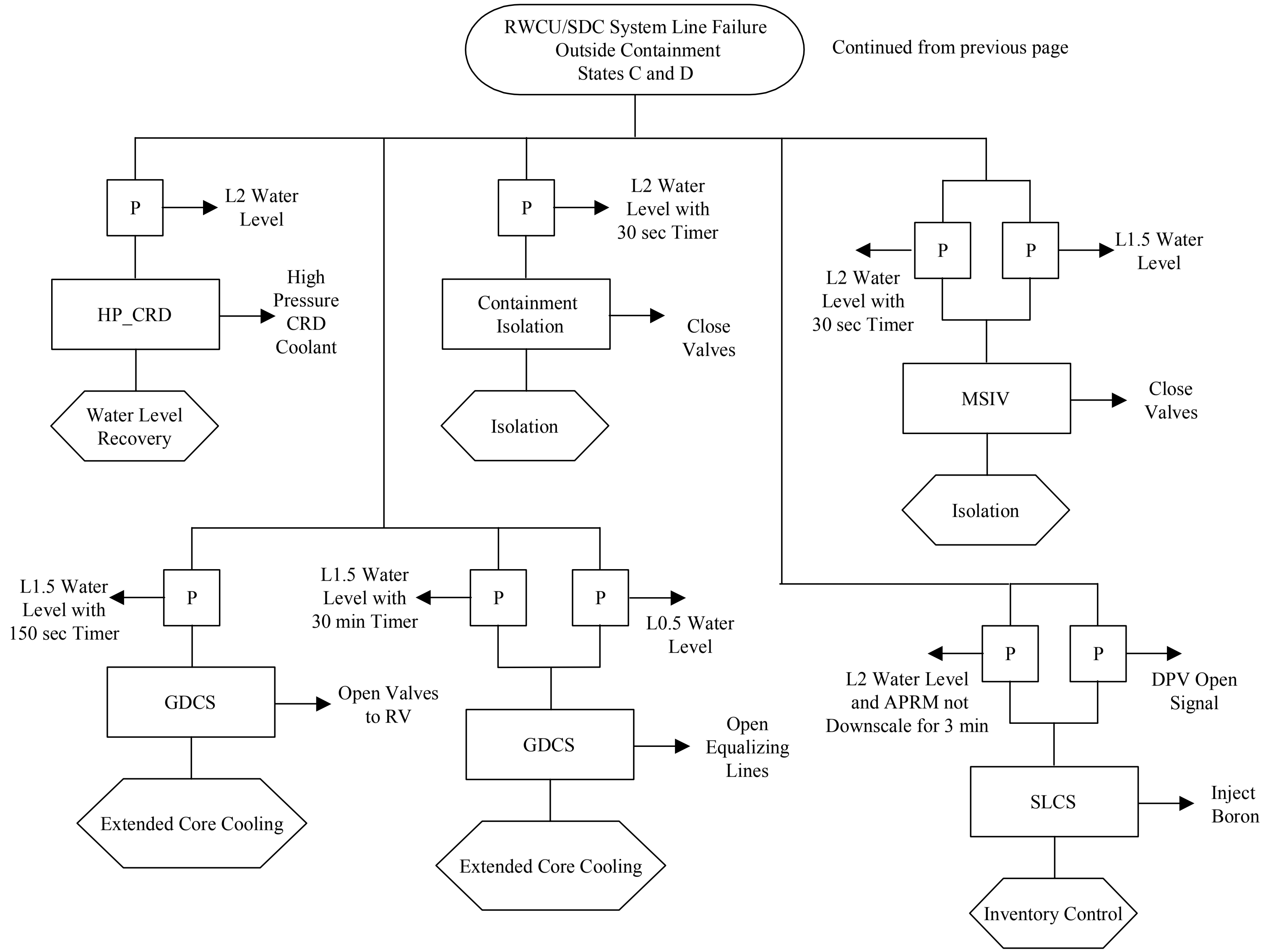


Figure 15.1-42. Event Diagram – Spent Fuel Cask Drop Accident

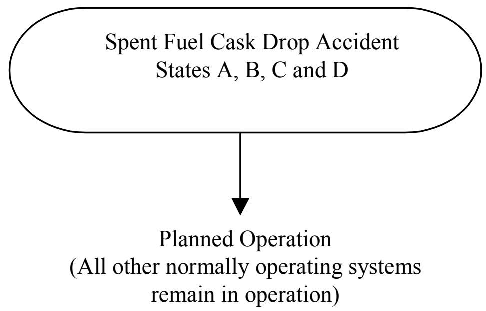


Figure 15.1-43. Event Diagram – MSIV Closure With Flux Scram (Overpressure Protection)

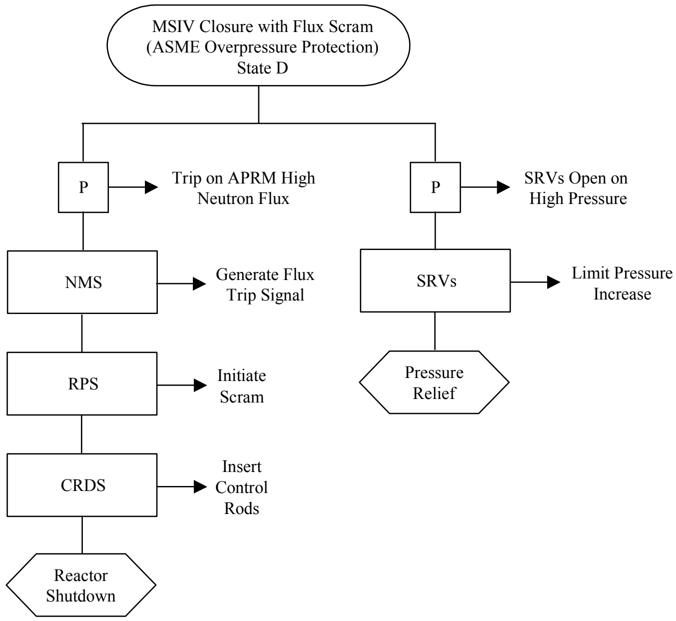


Figure 15.1-44. Event Diagram – Shutdown Without Control Rods (Standby Liquid Control System Capability)

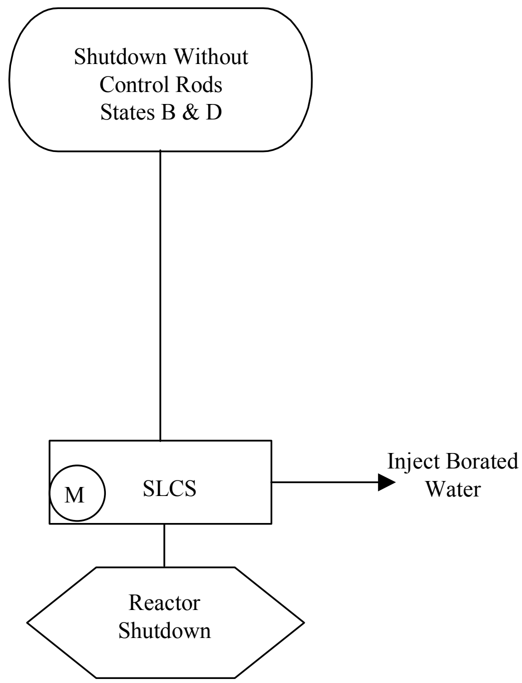


Figure 15.1-45. Event Diagram – Shutdown from Outside Main Control Room

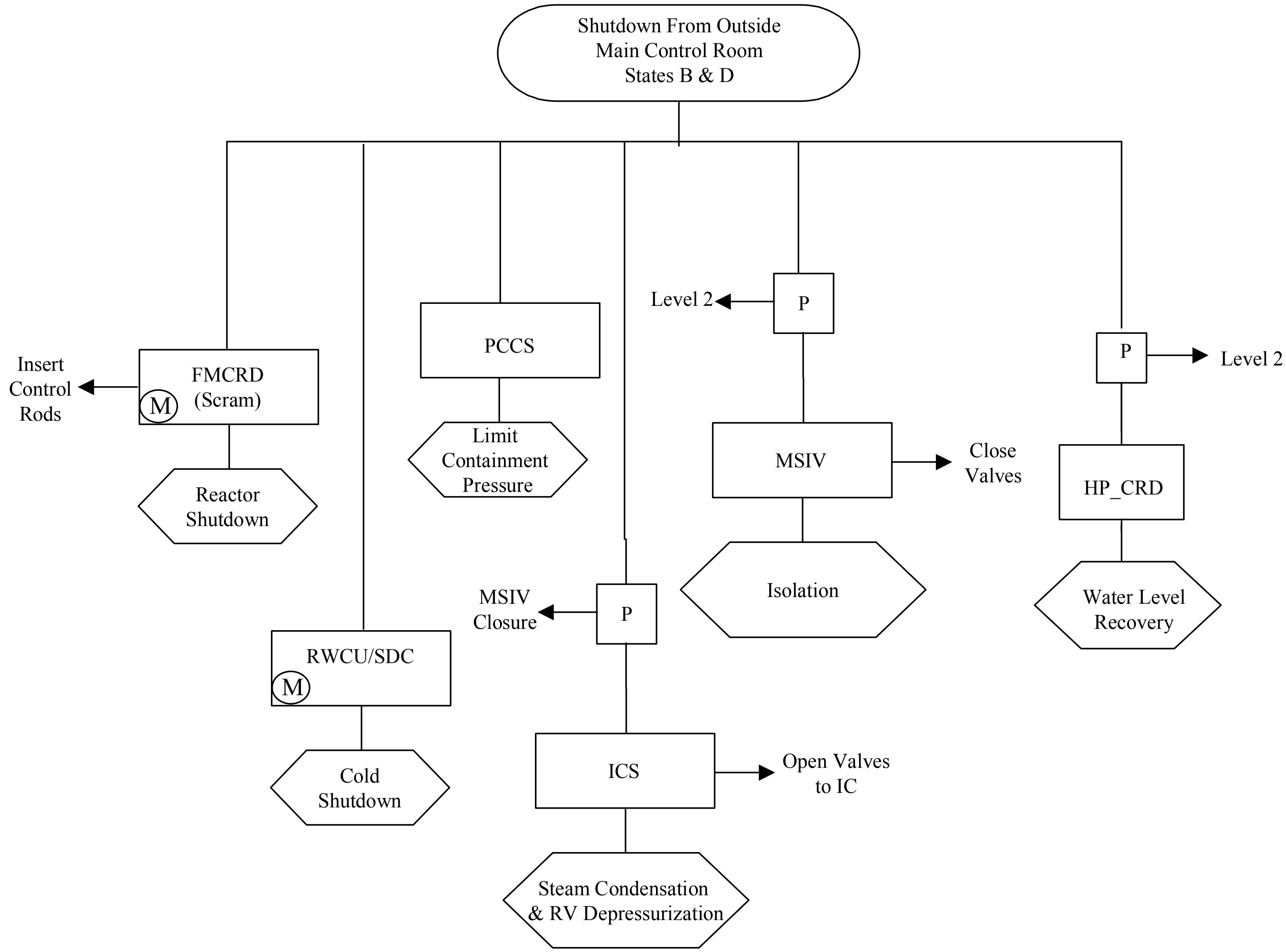


Figure 15.1-46a. Event Diagram – Anticipated Transients Without Scram

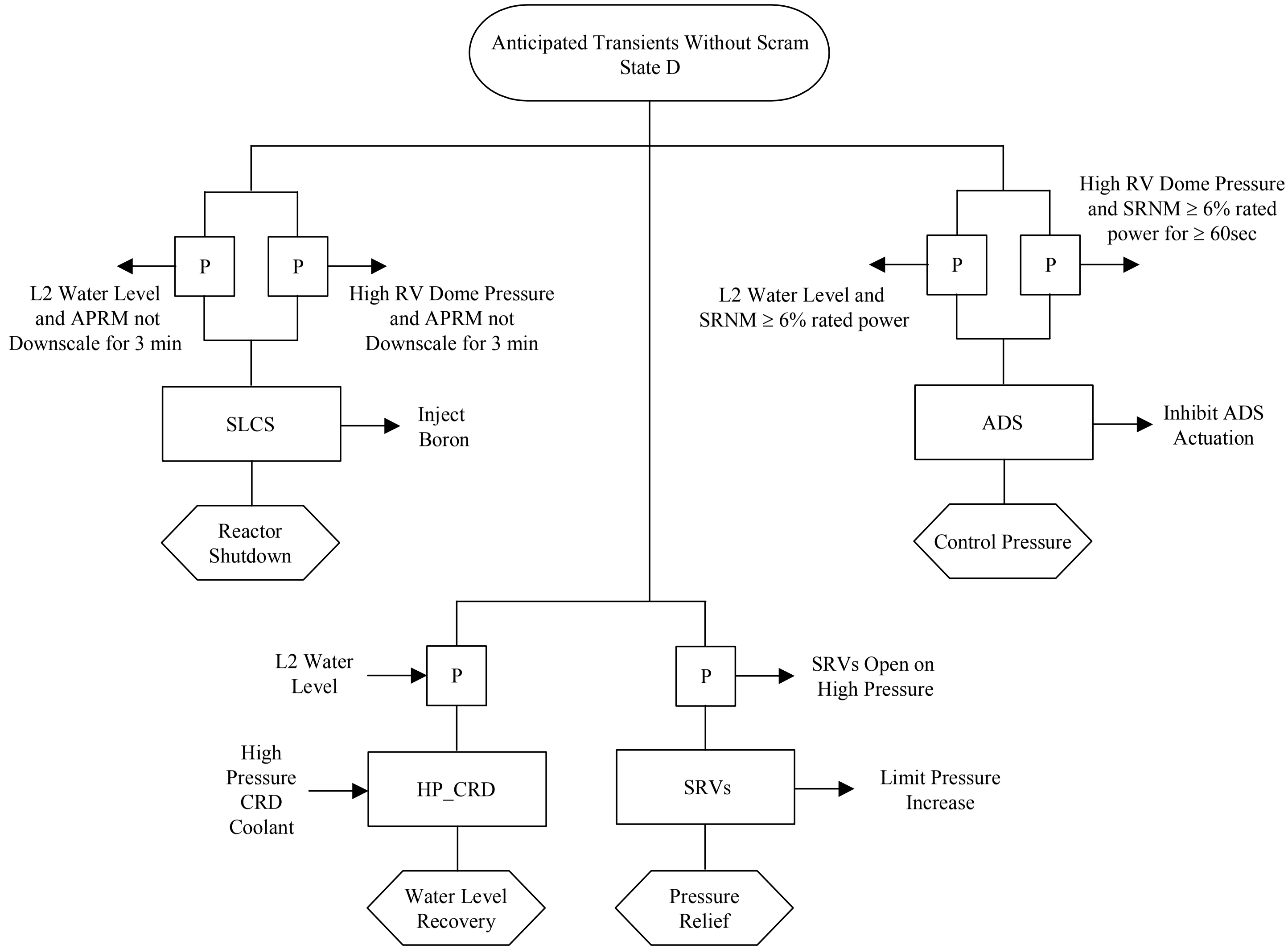


Figure 15.1-46b. Event Diagram – Anticipated Transients Without Scram

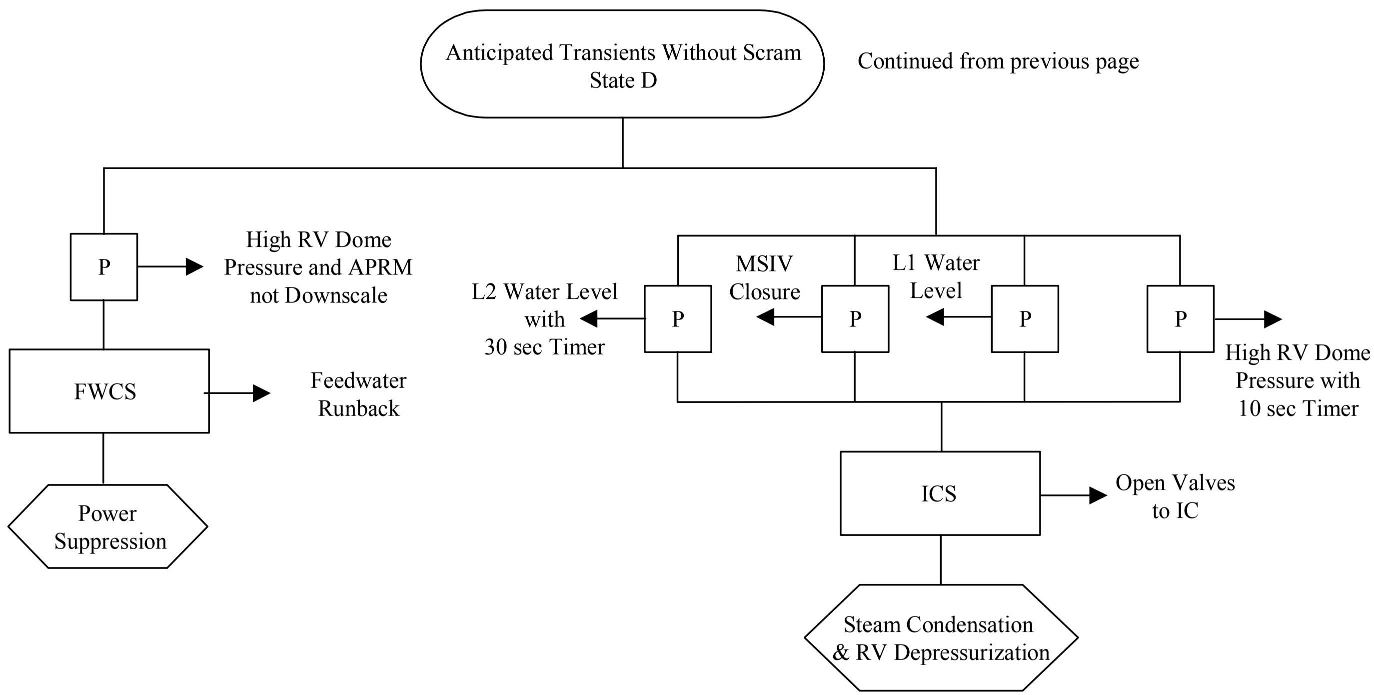


Figure 15.1-47a. Event Diagram – Station Blackout

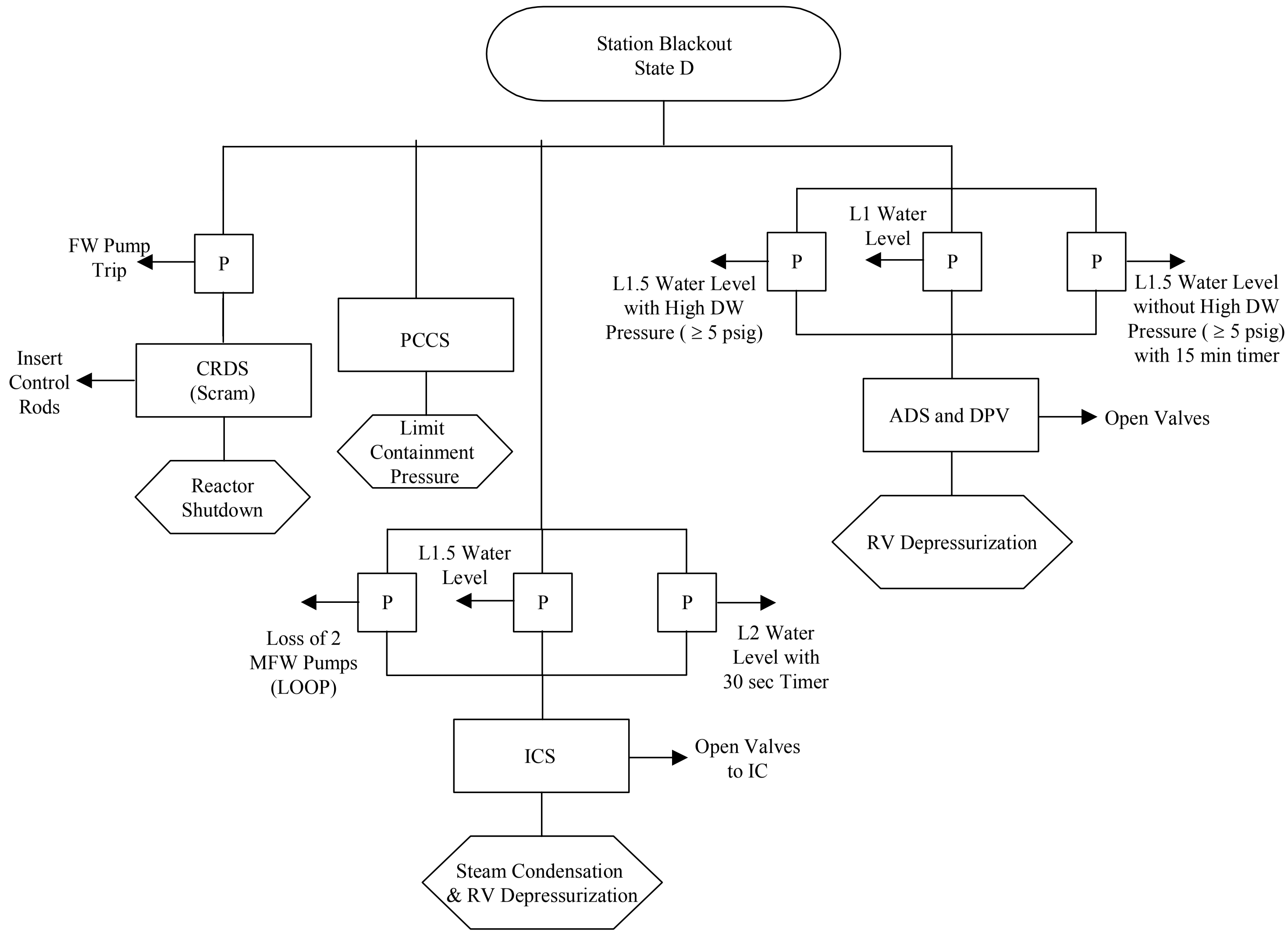


Figure 15.1-47b. Event Diagram – Station Blackout

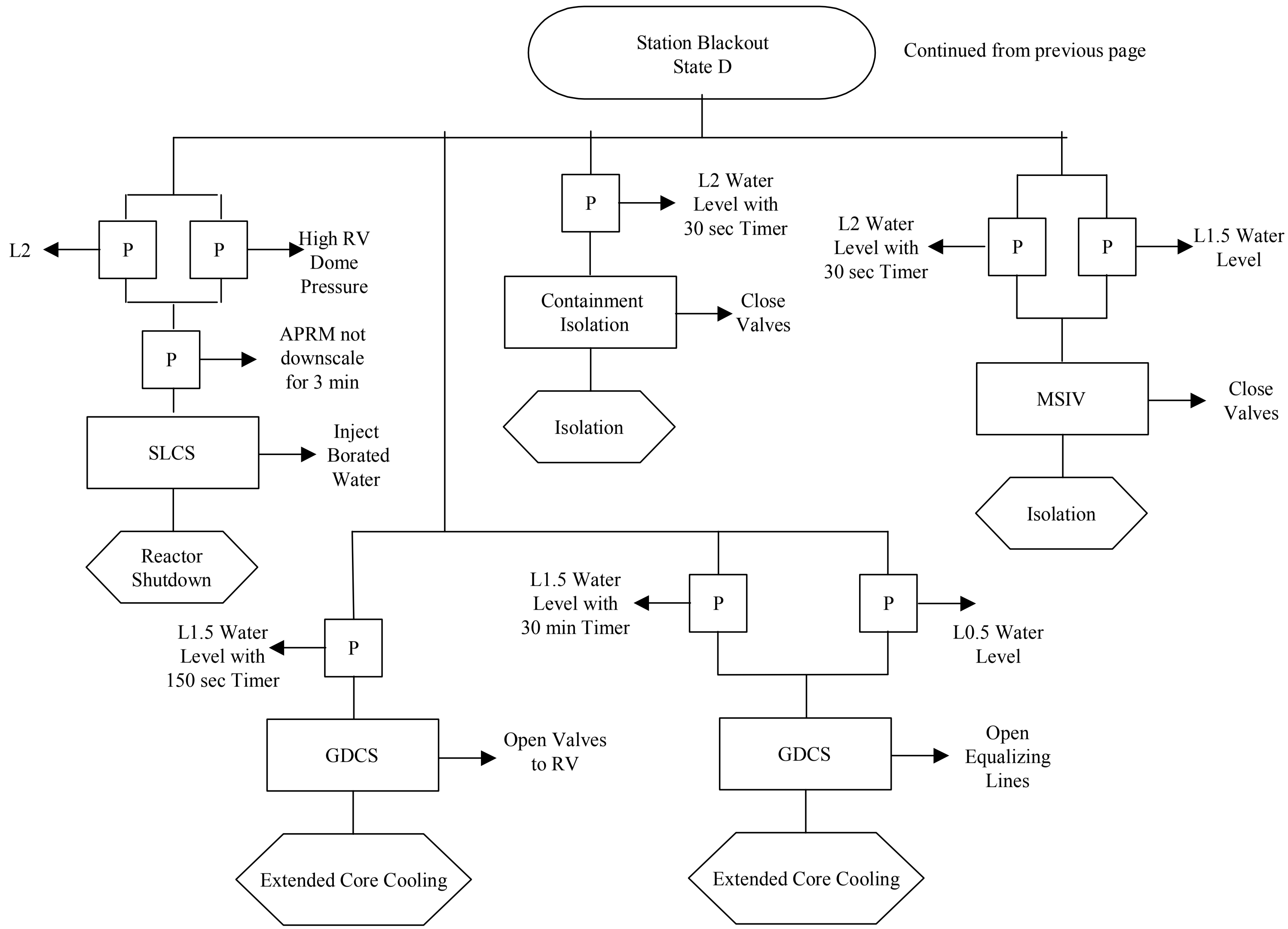


Figure 15.1-48. Event Diagram – Safe Shutdown Fire

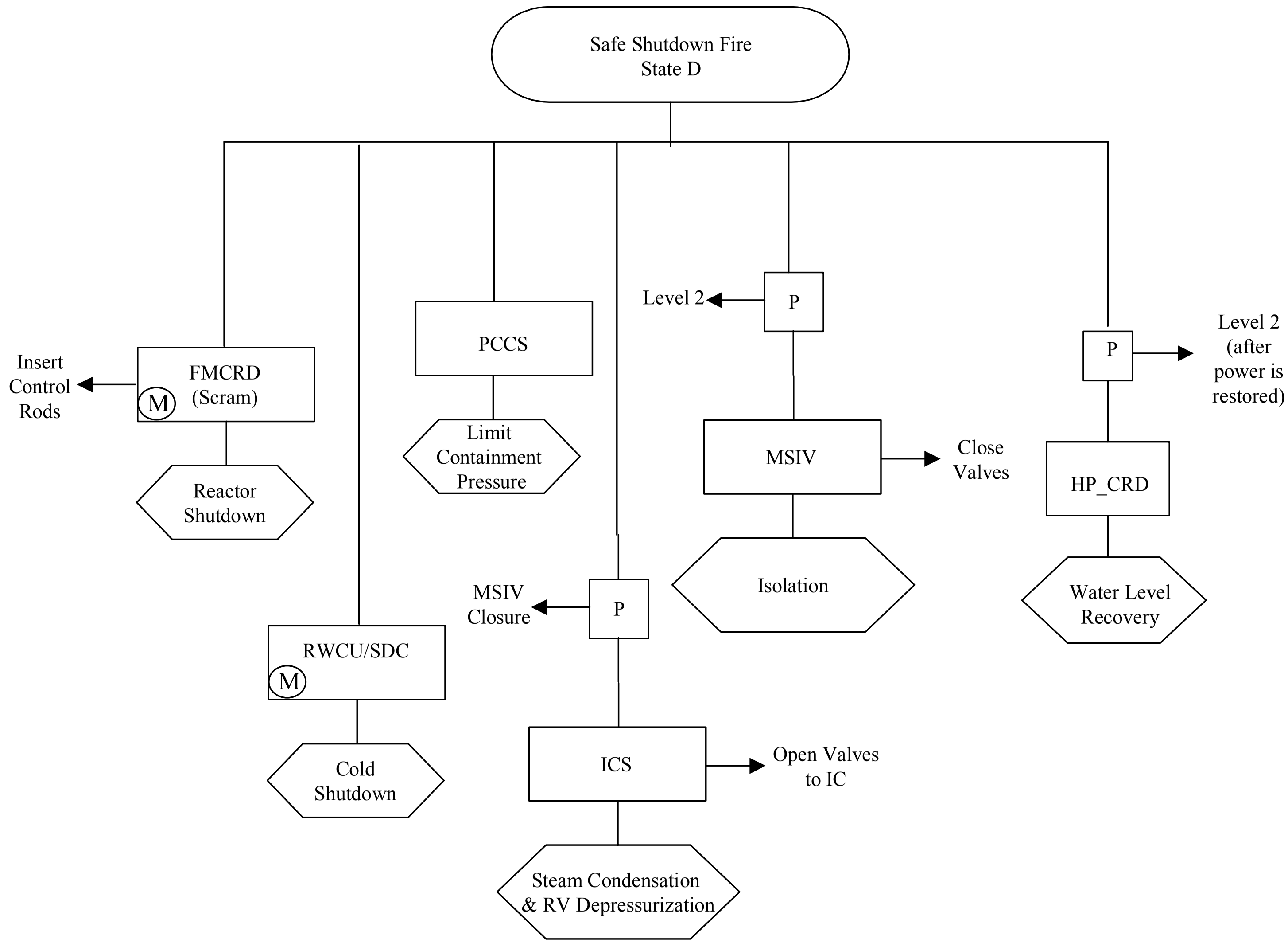
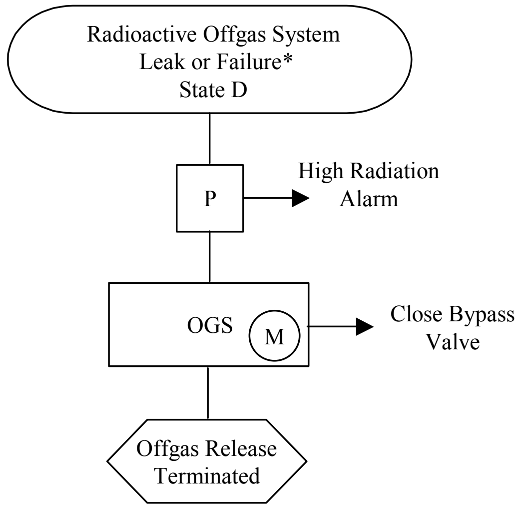


Figure 15.1-49. Event Diagram – Waste Gas System Leak or Failure



* Leak is assumed to result from inadvertent opening of a valve bypassing the charcoal adsorber tanks

15.2 ANALYSIS OF ANTICIPATED OPERATIONAL OCCURRENCES

Each of the anticipated operational occurrences (AOOs) addressed in the Section 15.1, "Nuclear Safety Operations Analysis" (NSOA), is evaluated in the following subsections. Tables 15.2-1, 15.2-2 and 15.2-3 provide the important input parameters and initial conditions used/assumed in the AOO analyses.

Assumptions

The following assumptions are applied in the TRACG calculations in Sections 15.2 and 15.3:

- The assumed initial suppression pool temperature is 43.3ºC (110ºF) and the scram set-point 48.9ºC (120ºF) in the inadvertent safety/relief valve (SRV) opening event analysis.
- A bounding isolation condenser (IC) injection valve stroke time of 0 seconds is assumed in the inadvertent IC injection analysis. Injection of all ICs (4 ICs) is assumed. The temperature of the condensate in the IC system that is initially injected is assumed to be 40ºC (104ºF).
- The turbine steam bypass system provides 50% flow in the event of a single failure.
- The feedwater controller failure analysis assumes a L8 feedwater runback signal. The L8 signal is backed up by a Safety Grade feedwater trip at L9.
- The feedwater control system provides 240s of rated feedwater flow (2434 kg/s) after main steam isolation valve (MSIV) isolation.
- The Loss of feedwater heating setpoint is assumed to be 16.67ºC (30ºF) measured in the feedwater nozzle to maximize the SCRRI actuation delay.
- The automatic depressurization system (ADS) low water level setpoint analytical limit is 13 m above vessel 0, in combination with high drywell pressure (for LOCA), and 11 m above vessel 0 without high DW pressure.
- The maximum feedwater pump runout for a single pump is 75% of rated flow.
- For the transient representative of the loss of off site power with failure to transfer to internal power sources, it is assumed that initially a load rejection occurs, feedwater pumps trip and condensate pumps trip simultaneously.

15.2.1 Decrease In Core Coolant Temperature

15.2.1.1 Loss Of Feedwater Heating

15.2.1.1.1 Identification of Causes

A feedwater (FW) heater can be lost in at least two ways:

- steam extraction line to heater is closed; or
- FW is bypassed around heater.

The first case produces a gradual cooling of the FW. In the second case, the FW bypasses the heater and no heating of the FW occurs. In either case, the reactor vessel receives colder FW.

The maximum number of FW heaters that can be tripped or bypassed by a single event represents the most severe event for analysis considerations.

The ESBWR is designed such that no single operator error or equipment failure shall cause a loss of more than 55.6°C (100°F) FW heating. The reference steam and power conversion system shown in Section 10.1 meets this requirement. In fact, the FW temperature drop based on the reference heat balance shown in Section 10.1 is less than 39°C (70°F). Therefore, the analyzed FW temperature drop shown in Table 15.2-1 is conservative.

The loss of FW heating causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient. However, the power increase is slow.

The Feedwater Control System (FWCS) logic is provided in Subsection 7.7.3, and includes logic to mitigate the effects of a loss of FW heating capability. The system is constantly monitoring the actual FW temperature and comparing it with a reference temperature. When a loss of FW heating is detected [i.e., when the difference between the actual and reference temperatures exceeds a ΔT setpoint], the FWCS sends an alarm to the operator and sends a signal to the Rod Control and Information System (RC&IS) to initiate the selected control rods run-in (SCRRI) function to automatically reduce the reactor power and avoid a scram. This prevents the reactor from violating any thermal limits.

Control blade insertion is conservatively assumed to start only when the temperature difference setpoint is reached in the FW nozzle. If the core axial power shape is top-peaked, the SCRRI is not able to suppress totally the neutron power increase and the MCPR margin is reduced, and the ΔCPR reduction is maximized. The flux transient is turned around when the control blades, during their insertion reach the upper part of the core where the power has increased.

Credit is not taken for the High Flux and High Simulated Thermal Power scrams because the core power transient depends on the assumed conditions, and under some situations, and for milder Loss of Feedwater Heating events, the setpoints for these scrams may not be reached.

15.2.1.1.2 Sequence of Events and Systems Operation

Sequence of Events

Table 15.2-4 lists the sequence of events for Figure 15.2-1

Because no scram occurs during this event, no immediate operator action is required. As soon as possible, the operator should verify that no operating limits are being exceeded. Also, the operator should determine the cause of failure prior to returning the system to normal.

Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection and reactor protection systems.

The high simulated thermal power trip (STPT) scram is the primary protection system trip in mitigating the effects of this event. However, the power increase in this event will not be high enough to guarantee the initiation of this scram for lower magnitude decreases in FW temperature.

15.2.1.1.3 Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by programming a change in FW enthalpy corresponding to the assumed loss in FW heating, shown in Table 15.2-1.

Results

Because the power increase during this event is relatively slow, although the reactor power does not reach high values, the core thermal flux increase can be relatively high, which can result in an appreciable reduction of the MCPR that is turned around when the SCRRI function takes effect. The results are summarized in Table 15.2-5.

No scram is assumed in this analysis. The increased core inlet subcooling aids thermal margins. Nuclear system pressure does not significantly change [< 0.03 MPa (5 psi)], and consequently, the RCPB is not threatened.

This sets the OLMCPR and will be reanalyzed for each core design and SCRRI rod pattern, for the cycle 0 core and reload cores. The COL applicant will provide a reanalysis of this event for the specific initial and reload core designs.

15.2.1.1.4 Barrier Performance

As noted previously, the effects of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed. Therefore, these barriers maintain their integrity and function as designed.

15.2.1.1.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.2 Increase In Reactor Pressure

15.2.2.1 Closure of One Turbine Control Valve

15.2.2.1.1 Identification of Causes

The ESBWR Steam Bypass and Pressure Control (SB&PC) system uses a triplicated digital control system. This system is similar to the one used in the ABWR design. The SB&PC system controls the TCVs and turbine bypass valves to maintain reactor pressure. As presented in Subsection 15.2.4.2, no credible single failure in the control system results in a minimum demand to all Turbine Control Valves (TCVs) and bypass valves. A voter or actuator failure may result in an inadvertent closure of one turbine control valve or one turbine bypass valve if it is open at the time of failure. In this case, the SB&PC system senses the pressure change and commands the remaining control valves or bypass valves, if needed, to open, and thereby automatically mitigates the transient to maintain reactor power and pressure.

Because turbine bypass valves are normally closed during normal full power operation, it is assumed for purposes of this transient analysis that a single failure causes a single turbine control valve to fail closed. Should this event occur at full power, the opening of remaining control valves may not be sufficient to maintain the reactor pressure, depending on the turbine design.

Neutron flux would increase in this case, due to void collapse resulting from the pressure increase. A reactor scram would be initiated if the high flux or high pressure scram setpoint is reached.

15.2.2.1.2 Sequence of Events and System Operation

Sequence of Events

Postulating an actuator failure of the SB&PC system causes one TCV to close. The pressure increases because the reactor is still generating the initial steam flow. The SB&PC system opens the remaining control valves and some bypass valves. This sequence of events is listed in Table 15.2-6 for Figure 15.2-2, for a fast closure with partial arc, and in Table 15.2-7 for Figure 15.2-3, for a slow closure with partial arc.

Systems Operation

Normal plant instrumentation and control are assumed to function. After a closure of one turbine control valve, the steam flow rate that can be transmitted through the remaining three TCVs depends upon the turbine configuration. For plants with full-arc turbine admission, the steam flow through the remaining three TCVs is at least 95% of rated steam flow. This capacity drops to about 85% of rated steam flow for plants with partial-arc turbine admission. Therefore, this transient is less severe for plants with full-arc turbine admission. In this analysis, the case with partial-arc turbine admission is analyzed to cover all plants.

Table 15.2-1 provides the following data for the TCV:

- Design full stroke closure time, from fully open to fully closed;
- Bounding closure time assumed in the fast closure analysis;
- Closure time assumed in the slow closure analysis; and
- Percent of rated steam flow that can pass through the three TCVs.

15.2.2.1.3 Core and System Performance

A simulated fast closure of one TCV is presented in Figure 15.2-2. Neutron flux increases, because of the void reduction caused by the pressure increase. However, the sensed neutron flux does not reach the high neutron flux scram setpoint. When the sensed reactor pressure increases, the pressure regulator opens the bypass valves, keeping the reactor pressure at a constant level. The calculated peak thermal flux is provided in Table 15.2-5. The number of rods in boiling transition for this transient remains within the acceptance criterion for AOOs. Therefore, the design basis is satisfied.

A slow closure of one TCV is also analyzed as shown in Figure 15.2-3. As in the fast closure case, the neutron flux increase does not reach the high neutron flux scram setpoint. Also, a reactor scram on high reactor pressure may also be generated. The results of this event are very similar to the fast closure event discussed above. During the transient, the number of rods in boiling transition remains within the acceptance criterion for AOOs.Therefore, the design basis is satisfied.

The COL applicant will provide a reanalysis of this event for the specific initial core configuration.

15.2.2.1.4 Barrier Performance

Peak absolute pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. The peak vessel bottom absolute pressure is below the upset pressure limit.

15.2.2.1.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.2.2 Generator Load Rejection With Turbine Bypass

15.2.2.2.1 Identification of Causes

Fast closure of the TCVs is initiated whenever electrical grid disturbances occur which result in significant loss of electrical load on the generator. The TCVs are required to close as rapidly as possible to prevent excessive over-speed of the turbine-generator (TG) rotor. Closure of the TCVs causes a sudden reduction in steam flow. To prevent an increase in system pressure, sufficient bypass capacity is provided to pass steam flow diverted from the turbine.

After sensing a significant loss of electrical load on the generator, the TCVs are commanded to close rapidly. At the same time, the turbine bypass valves are signaled to open in the "fast" opening mode by the SB&PC system, which uses a triplicated digital controller. As presented in Subsection 15.2.4.3, no single failure can cause all turbine bypass valves to fail to open on demand.

Assuming no single failure the plant will have the full steam bypass capability available, the Reactor Protection System (RPS) will verify that the bypass valves are open. The fast closure of the TCVs will produce a pressure increase that will be negligible, however, because all the steam flow will be bypassed through the steam bypass valves. The reactor will continue operating at the same steady state level until the SCRRI is able to begin reducing the power, as specified in the functional requirements of this system.

15.2.2.2.2 Sequence of Events and System Operation

Sequence of Events

A loss of generator electrical load from high power conditions produces the sequence of events listed in Table 15.2-8.

Identification of Operator Actions

Relatively small changes in plant conditions are experienced. The operator should, after checking that the SCRRI system has been activated, check reactor water level, reactor pressure and MSIV status. If conditions are normal, no further operator actions are needed.

System Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems unless stated otherwise.

All plant control systems maintain normal operation unless specifically designated to the contrary.

15.2.2.2.3 Core and System Performance

Input Parameters and Initial Conditions

The turbine electro-hydraulic control system (EHC) detects load rejection before a measurable turbine speed change takes place.

The closure characteristics of the TCVs are assumed such that the valves operate in the full arc (FA) mode. For this event, Table 15.2-1 provides the worst case full stroke closure time (from fully open to fully closed) for the TCVs, which is assumed in the analysis.

The bypass valve opening characteristics are simulated using the specified delay together with the specified opening characteristic required for bypass system operation.

Results

Figure 15.2-4 shows the results of the generator trip from the 100% rated power conditions and with the turbine bypass system operating normally. Although the peak neutron flux and average simulated thermal heat flux increase**,** the number of rods expected in boiling transition remains within the acceptance criterion for AOOs.. This event will be reanalyzed for each specific initial core configuration.

15.2.2.2.4 Barrier Performance

Peak absolute pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. The peak vessel bottom absolute pressure remains below the upset pressure limit.

15.2.2.2.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.2.3 Generator Load Rejection With a Single Failure in the Turbine Bypass System

15.2.2.3.1 Identification of Causes

Fast closure of the TCVs is initiated whenever electrical grid disturbances occur which result in significant loss of electrical load on the generator. The TCVs are required to close as rapidly as possible to prevent excessive overspeed of the turbine-generator (TG) rotor. Closure of the TCVs causes a sudden reduction in steam flow, which results in an increase in system pressure and reactor shutdown if the available turbine steam bypass capacity is insufficient.

After sensing a significant loss of electrical load on the generator, the TCVs are commanded to close rapidly. At the same time, the turbine bypass valves are signaled to open in the "fast" opening mode by the SB&PC system, which uses a triplicated digital controller. As presented in Subsection 15.2.4.3, no single failure can cause all turbine bypass valves to fail to open on demand. It is assumed that half of the turbine bypass valves fail to open on demand in this analysis.

15.2.2.3.2 Sequence of Events and System Operation

Sequence of Events

A loss of generator electrical load with a single failure in the turbine bypass system from high power conditions produces the sequence of events listed in Table 15.2-9.

Identification of Operator Actions

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Verify proper bypass valve performance;
- Observe that the FW/level controls have maintained the reactor water level at a satisfactory value;
- Observe that the pressure regulator is controlling reactor pressure at the desired value; and
- Observe reactor peak power and pressure.

System Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems unless stated otherwise.

Conservatively, and to cover all possible failures, it is assumed that the system with a single failure only opens to 50% of the total steam bypass capacity.

All plant control systems maintain normal operation unless specifically designated.

15.2.2.3.3 Core and System Performance

Input Parameters and Initial Conditions

The turbine electro-hydraulic control system (EHC) detects load rejection before a measurable turbine speed change takes place.

The closure characteristics of the TCVs are assumed such that the valves operate in the full arc (FA) mode. For this event, Table 15.2-1 provides the design full stroke closure time (from fully open to fully closed) for the TCVs and the worst-case closure time is assumed in the analysis.

The bypass valve opening characteristics are simulated using the specified delay together with the specified opening characteristic required for bypass system operation.

The pressurization and/or the reactor scram may compress the water level to the low level trip setpoint (Level 2) and initiate the CRD high pressure makeup function, and if the low level signal remains for 30 seconds, MSIV closure, and isolation condenser (IC) operation. Should this occur, it would follow sometime after the primary concerns of fuel thermal margin and overpressure effects have occurred.

Results

Figure 15.2-5 shows the results of the generator trip from the 100% rated power conditions assuming only 50% of the total turbine bypass system capacity. Although the peak neutron flux and average simulated thermal heat flux increase, the number of rods in boiling transition remains within the acceptance criterion for AOOs. This event will be reanalyzed for each specific initial core configuration.

15.2.2.3.4 Barrier Performance

Peak absolute pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. The peak vessel bottom absolute pressure remains below the upset pressure limit.

15.2.2.3.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.2.4 Turbine Trip With Turbine Bypass

15.2.2.4.1 Identification of Causes

A variety of turbine or nuclear system malfunctions can initiate a turbine trip. Some examples are high velocity separator drain tank high levels, large vibrations, operator lockout, loss of control fluid pressure, low condenser vacuum and reactor high water level.

After the main turbine is tripped, turbine bypass valves are opened in their fast opening mode by the SB&PC system.

15.2.2.4.2 Sequence of Events and Systems Operation

Sequence of Events

Turbine trip at high power produces the sequence of events listed in Table 15.2-10.

Identification of Operator Actions

Relatively small changes in plant conditions are experienced. The operator should, after checking that the SCRRI system has been activated, check reactor water level, reactor pressure and MSIV status. If conditions are normal, no further operator actions are needed.

Systems Operation

All plant control systems maintain normal operation unless specifically designated to the contrary. Credit is taken for successful operation of the Reactor Protection System (RPS).

15.2.2.4.3 Core and System Performance

Input Parameters and Initial Conditions

Table 15.2-1 provides the Turbine Stop Valve (TSV) full stroke closure time design range, and the worst case (bounding) TSV closure time assumed in the analysis.

Results

A turbine trip with the bypass system operating normally is simulated at rated power conditions as shown in Figure 15.2-6. Table 15.2-5 summarizes the analysis results. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the pressure increase is limited by the initiation of the steam bypass operation. Peak simulated thermal heat flux does not significantly exceed (< 1%) of its initial value. After the control system verifies that the bypass capacity is adequate, the system will activate the SCRRI to reduce the power to 60% and later proceed to a possible restart or a controlled shut-down. The number of rods in boiling transition during this event remains within the acceptance criterion for AOOs. This event will be reanalyzed for each specific initial core configuration.

15.2.2.4.4 Barrier Performance

Peak absolute pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. The peak absolute pressure at the vessel bottom remains below the upset pressure limit.

15.2.2.4.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.2.5 Turbine Trip With a Single Failure in the Turbine Bypass System

15.2.2.5.1 Identification of Causes

A variety of turbine or nuclear system malfunctions can initiate a turbine trip. Some examples are high velocity separator drain tank high levels, large vibrations, operator lockout, loss of control fluid pressure, low condenser vacuum and reactor high water level.

After the main turbine is tripped, turbine bypass valves are opened in their fast opening mode by the SB&PC system.

15.2.2.5.2 Sequence of Events and Systems Operation

Sequence of Events

Turbine trip with a single failure in the turbine bypass system at high power produces the sequence of events listed in Table 15.2-11.

Identification of Operator Actions

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Verify that the generator breaker is automatically open to allow electrical buses originally supplied by the generator to be supplied by the incoming power;
- Monitor reactor water level and pressure;
- Check turbine for proper operation of all auxiliaries during coastdown;

- Manually initiate ICs, if necessary, to control reactor pressure;
- Depending on conditions, maintain pressure for restart purposes, or initiate normal operating procedures for cooldown;
- Put the mode switch in the startup position before the reactor pressure decays to below 6 MPa (870 psig);

Systems Operation

All plant control systems maintain normal operation unless specifically designated to the contrary. Credit is taken for successful operation of the Reactor Protection System (RPS).

Conservatively and to cover all possible failures it is assumed that the system with a single failure only opens to 50% of the total steam bypass capacity.

15.2.2.5.3 Core and System Performance

Input Parameters and Initial Conditions

Table 15.2-1 provides the Turbine Stop Valve (TSV) full stroke closure time design range, and the worst case (bounding) TSV closure time assumed in the analysis. A reactor scram occurs due to fast TSV closure, with inadequate availability of turbine bypass.

Results

A turbine trip, assuming only 50% of the total turbine steam bypass capacity available, is simulated at rated power conditions as shown in Figure 15.2-7. Table 15.2-5 summarizes the analysis results. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the flux increase is limited by the partial actuation of the steam bypass system and the initiation of reactor scram. The peak simulated thermal heat flux does not significantly increase (< 10%) above its initial value. The number of rods in boiling transition during this event remains within the acceptance criterion for AOOs. This event will be reanalyzed for each specific initial core configuration.

15.2.2.5.4 Barrier Performance

Peak absolute pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. The peak absolute pressure at the vessel bottom remains below the upset pressure limit.

15.2.2.5.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.2.6 Closure of One Main Steamline Isolation Valve

15.2.2.6.1 Identification of Causes

Protection system logic permits the test closure of one MSIV without initiating scram from the position switches. An inadvertent closure of one MSIV may cause an immediate closure of all other MSIVs, depending on reactor conditions. Closure of all MSIVs is discussed in Subsection 15.2.2.7.

15.2.2.6.2 Sequence of Events and Systems Operation

When a single MSIV is closed in conformance with normal testing procedures, no reactor scram occurs and the reactor settles into a new steady state operating condition. Closure of a single MSIV at power levels above those of the normal testing procedure may cause closure of all other MSIVs.

Table 15.2-12 lists the sequence of events for Figure 15.2-8

15.2.2.6.3 Core and System Performance

The neutron flux increases slightly while the simulated thermal heat flux shows no increase. The number of rods in boiling transition during this event remains within the acceptance criterion for AOOs. The effects of closure of a single MSIV are considerably milder than the effects of closure of all MSIVs. Therefore, this event does not need to be reanalyzed for any specific core configuration.

Inadvertent closure of one MSIV while the reactor is shut down produces no significant transient. Closures during plant heatup are less severe than closure from maximum power cases.

15.2.2.6.4 Barrier Performance

Peak absolute pressure at the vessel bottom remains below the pressure limits of the reactor coolant pressure boundary. Peak absolute pressure in the main steamline remains below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool.

15.2.2.6.5 Radiological Consequence

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.2.7 Closure of All Main Steamline Isolation Valves

15.2.2.7.1 Identification of Causes

Various steamline and nuclear system malfunctions, or operator actions, can initiate MSIV closure. Examples are low steamline pressure, high steamline flow, high steamline radiation, low water level or manual action.

To define this event as an initiating event and not the byproduct of another AOO, only the following are considered:

- Manual action (purposely or inadvertent);
- Spurious signals such as low pressure, low reactor water level, low condenser vacuum; and
- Equipment malfunctions, such as faulty valves or operating mechanisms.

A closure of one MSIV may cause an immediate closure of all other MSIVs, depending on reactor conditions. If this occurs, it is also included in this category. During the MSIV closure, position switches on the valves provide a reactor scram if the valves in two or more main steamlines are less than that shown in Table 15.2-1 (except for interlocks which permit proper plant startup). Protection system logic, however, switches to two out of three of the remaining

MSIVs which permits the test closure of one valve without initiating scram from the position switches.

15.2.2.7.2 Sequence of Events and Systems Operation

Sequence of Events

Table 15.2-13 lists the sequence of events for Figure 15.2-9.

The following is the sequence of operator actions expected during the course of the event, assuming no restart of the reactor. The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Check that ICs have initiated (i.e., drain valves open);
- Monitor reactor water level and pressure;
- Initiate RWCU/SDC system operation in the shutdown cooling mode appropriate to hot shutdown;
- Determine the cause of valve closure before resetting the MSIV isolation;
- Observe turbine coastdown and break vacuum before the loss of sealing steam (check turbine auxiliaries for proper operation); and
- Check that conditions are satisfactory prior to opening and resetting MSIVs.

Systems Operation

MSIV closure initiates a reactor scram trip via position signals to the RPS. The same signal also initiates the operation of ICs, which prevent the lifting of SRVs.

All plant control systems maintain normal operation unless specifically designated to the contrary.

15.2.2.7.3 Core and System Performance

Input Parameters and Initial Conditions

The MSIV design closure time range and the worst case (bounding) closure time assumed in this analysis are provided in Table 15.2-1.

Position switches on the valves initiate a reactor scram, as addressed in Table 15.2-1. Closure of these valves causes the dome pressure to increase.

Results

Figure 15.2-9 shows the changes in important nuclear system variations for the simultaneous isolation of all main steamlines while the reactor is operating at rated power. The neutron flux increases slightly while the simulated thermal heat flux shows no increase. The FW injection and the IC operation terminate the pressure increase. The anticipatory scram prevents any change in the thermal margins. The number of rods in boiling transition during this event remains within the acceptance criterion for AOOs. Therefore, this event does not have to be reanalyzed for any specific core configurations.

Inadvertent closure of all of the MSIVs while the reactor is shut down produces no significant transient. Closures during plant heatup are less severe than the maximum power cases (maximum stored and decay heat) presented.

15.2.2.7.4 Barrier Performance

Peak absolute pressure at the vessel bottom remains below the upset event pressure limit for the reactor coolant pressure boundary (RCPB). Peak absolute pressure in the main steamline remains below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool.

15.2.2.7.5 Radiological Consequence

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.2.8 Loss of Condenser Vacuum

15.2.2.8.1 Identification of Causes

Various system malfunctions that can cause a loss of condenser vacuum due to some single equipment failure are designated in Table 15.2-14.

15.2.2.8.2 Sequence of Events and Systems Operation

Sequence of Events

Table 15.2-15 lists the sequence of events for Figure 15.2-10.

The Loss of Condenser Vacuum initially does not effect the vessel, when the turbine trip setpoint is reached it has a simultaneous scram with a bypass valve opening. According to Table 15.2-16, six seconds later the low vacuum setpoint produces closure of the bypass and with a small delay the MSIV also closes.

Identification of Operator Actions

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Monitor reactor water level and pressure;
- Check turbine for proper operation of all auxiliaries during coastdown;
- Use ICs to control reactor pressure;
- Depending on conditions, maintain pressure for restart purposes, or initiate normal operating procedures for cooldown;
- Put the mode switch in the STARTUP position before the reactor pressure decays below 6 MPa (870 psig).

Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection and reactor protection systems.

Tripping functions incurred by sensing main turbine condenser vacuum are presented in Table 15.2-16.

15.2.2.8.3 Core and System Performance

Input Parameters and Initial Conditions

TSV full stroke closure time is as shown in Table 15.2-1.

A reactor scram is initiated on low condenser vacuum at the same time that the turbine trip signal is generated.

The analysis presented here is a hypothetical case with a conservative vacuum decay rate (see Table 15.2-1). Thus, the bypass system is available for several seconds, because the bypass is signaled to close at a vacuum level that is less than the stop valve closure (see Table 15.2-16).

Results

As shown in Table 15.2-15, under the analysis vacuum decay condition, the turbine bypass valves and MSIV closure would follow main turbine trip after it initiates the event. This AOO, therefore, is similar to a normal turbine trip with bypass. The effect of MSIV closure tends to be minimal, because the reactor scram on low condenser vacuum precedes the isolation by several seconds. Figure 15.2-10 shows the transient expected for this event. It is assumed that the plant is initially operating at rated power conditions. Peak neutron flux is shown in Table 15.2-5, and the average simulated thermal heat flux peaks at < 110% of rated. The number of rods in boiling transition during this event remains within the acceptance criterion for AOOs. Therefore, this event does not have to be reanalyzed for any specific core configuration.

15.2.2.8.4 Barrier Performance

Peak nuclear system absolute pressure remains below the ASME code upset limit. Peak absolute pressure in the main steamline remains below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. A comparison of these values to those for turbine trip at high power shows the similarities between these two transients. The prime difference is the subsequent main steamline isolation.

15.2.2.8.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.2.9 Loss of Shutdown Cooling Function of RWCU/SDC

Although the RWCU/SDC system is not safety-related, it can perform high and low pressure core cooling. The RWCU/SDC system has two trains, each of which contains the necessary piping, pumps, valves, heat exchangers, instrumentation and electrical power for operation.Each train also has its own cooling water supply, connection to standby AC power, pump, and equipment room cooling system. For the shutdown cooling function each train has its own suction line from the RPV and return line to the FW line. Thus each of the RWCU/SDC trains is completely independent of the other train. The RWCU/SDC system, together with the main condenser, is used to reduce the primary system temperature after plant shutdown.

Normally, in evaluating component failure considerations associated with RWCU/SDC system shutdown cooling mode operation, active pumps, valves or instrumentation would be assumed to fail. If the single active failure criterion is applied to the RWCU/SDC system, one of the RWCU/SDC trains would be inoperable. However, the operable RWCU/SDC train could achieve cold shutdown to ≤ 100°C (212°F) within 36 hours after reactor shutdown.

Failure of offsite power is another case that could affect the shutdown cooling function. The plant has two independent offsite power supplies. If both offsite power supplies are lost, each RWCU/SDC train has its own standby AC power source (e.g., diesel generator) that permits operating that train at its rated capacity. Application of the single active failure criterion would still leave an RWCU/SDC train operational.

The RWCU/SDC system description and performance evaluation in Subsection 5.4.8 describes the models, assumptions and results for shutdown cooling with two RWCU/SDC trains operational.

15.2.3 Reactivity and Power Distribution Anomalies

There are no reactivity and power distribution anomaly AOOs identified for the ESBWR.

15.2.4 Increase in Reactor Coolant Inventory

15.2.4.1 Inadvertent Isolation Condenser Initiation

15.2.4.1.1 Identification of Causes

Manual startup of the four individual IC systems is postulated for this analysis (i.e., operator error).

15.2.4.1.2 Sequence of Events and System Operation

Sequence of Events

Table 15.2-17 lists the sequence of events for Inadvertent Isolation Condenser Initiation.

Identification of Operator Actions

Relatively small changes in plant conditions are experienced. The operator should, after hearing the alarm that the IC system has commenced operation, check reactor water level, reactor pressure and MSIV status. If conditions are normal, the operator should shut down the system.

System Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls. Specifically, the pressure regulation and the vessel level control that respond directly to this event.

Required operation of engineered safeguards other than what is described is not expected for this event.

15.2.4.1.3 Core and System Performance

Input Parameter and Initial Conditions

The assumed IC system water temperature and enthalpy, startup time and flow rate are provided in Table 15.2-1.

Inadvertent startup of all loops of the IC system was chosen as the limiting case for analysis because it provides the greatest auxiliary source of cold water into the vessel.

Results

Figure 15.2-11 shows the simulated transient event. It begins with the introduction of cold water into the downcomer region. Full IC loop flow is established. No delays are considered because these are not relevant to the analysis.

Addition of cooler water to the downcomer causes a reduction in inlet enthalpy, which results in a power increase. The flux level settles out slightly above its operating level. The variations in the pressure and thermal conditions are relatively small, and no significant effect is experienced. The number of rods in boiling transition remains within the acceptance criterion for AOOs, and the fuel thermal margins are maintained.

Consideration of Uncertainties

Important analytical factors, including IC loop condensate water temperature, are assumed to be at the worst conditions so that any deviations in the actual plant parameters would produce a less severe transient.

15.2.4.1.4 Barrier Performance

Inadvertent Startup of the IC causes only a slight pressure decrease from the initial conditions; therefore, no further RCPB pressure response evaluation is required.

15.2.4.1.5 Radiological Consequences

Because no activity is released during this event, a detailed evaluation is not required.

15.2.4.2 Runout of One Feedwater Pump

15.2.4.2.1 Identification of Causes

This event is postulated on the basis of a single failure of a control device, specifically one that can directly cause an increase in coolant inventory by increasing the FW flow.

The ESBWR FWCS uses a triplicated digital control system, instead of a single-channel analog system that was originally provided in current BWR designs (BWR/2-6). The digital systems consist of a triplicated fault-tolerant digital controller, the operator control stations and displays. The digital controller contains three parallel processing channels, each containing the microprocessor-based hardware and associated software necessary to perform all the control calculations. The operator interface provides information regarding system status and the required control functions.

Redundant transmitters are provided for key process inputs, and input voting and validation are provided such that faults can be identified and isolated. Each system input is triplicated internally and sent to the three processing channels (Figure 15.2-12). The channels produce the same output during normal operation. Interprocessor communication provides self-diagnostic capability. A two-out-of-three voter compares the processor outputs to generate a validated output to the control actuator. A separate voter is provided for each actuator. A "ringback" feature feeds back the final voter output to the processors. A voter failure is thereby detected and alarmed. In some cases, a protection circuit locks the actuator into its existing position promptly after the failure is detected.

Table 15.2-18 lists the potential failure modes of a triplicated digital control system and outlines the effects of each failure. Because of the triplicated architecture, it is possible to take one channel out of service for maintenance or repair while the system is online. Modes 2 and 5 of Table 15.2-18 address a failure of a component while an associated redundant component is out of service. This type of failure could potentially cause a system failure. However, the probability of a component failure during servicing of a counterpart component is considered to be so low that these failure modes are not considered Anticipated Operational Occurences, but are considered infrequent events.

Adverse effect minimization is mentioned in the effects of Mode 2. This feature stems from the additional intelligence of the system provided by the microprocessor. When possible, the system is programmed to take action in the event of some failure to reduce the severity of the event. For example, if the total steam flow or total FW flow signals fail, the FWCS detects this by the input reasonability checks and automatically switch to one-element mode (i.e., control by level feedback only). The level control would essentially be unaffected by this failure.

The only credible single failures that would lead to some adverse effect on the plant are Modes 6 (failure of the output voter) and 7 (control actuator failure). Either of these failures would lead to a loss of control of only one actuator (i.e., only one FW pump with increasing flow). A voter failure is detected by the ringback feature. The FWCS initiates a lockup of the actuator upon detection of the failure. The probabilities of failure of the variety of control actuators are very low based on operating experience. The worst single failure in the FWCS causes a runout of one FW pump to its maximum capacity. In the event of one pump run-out, the FWCS would then reduce the demand to the remaining pumps, thereby automatically compensating for the excessive flow from the failed pump. However, the demand to the remaining FW pump decreases to offset the increased flow of the failed pump. The effect on total flow to the vessel is not significant. The worst additional single failure would cause all FW pumps to run out to their maximum capacity. However, the probability of this occurrence is extremely low.

15.2.4.2.2 Sequence of Events and Systems Operation

Sequence of Events

With momentary increase in FW flow, the water level rises and then settles back to its normal level. Table 15.2-19 lists the sequencing of events for Figure 15.2-13.

Identification of Operator Actions

Because no scram occurs for runout of one FW pump, no immediate operator action is required. As soon as possible, the operator should verify that no operating limits are being exceeded. Also, the operator should determine the cause of failure prior to returning the system to normal.

Systems Operation

Runout of a single FW pump requires no protection system or ESF system operation. This analysis assumes normal functioning of plant instrumentation and controls.

15.2.4.2.3 Core and System Performance

Input Parameters and Initial Conditions

The total FW flow for all pumps runout is provided in Table 15.2-1.

Results

The simulated runout of one FW pump event is presented in Figure 15.2-13. When the increase of FW flow is sensed, the FW controller starts to command the remaining FW pump to reduce its flow immediately. The vessel water level increases slightly [about 14 cm (6 inch)] and then settles back to its normal level. Vessel pressure increases insignificantly, and the number of rods in boiling transition remains within the acceptance criterion for AOOs.

15.2.4.2.4 Barrier Performance

As previously noted, the effect of this event does not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed. Therefore, these barriers maintain their integrity and function as designed.

15.2.4.2.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.5 Decrease in Reactor Coolant Inventory

15.2.5.1 Opening of One Turbine Control or Bypass Valve.

15.2.5.1.1 Identification of Causes

The ESBWR Steam Bypass and Pressure Control (SB&PC) system uses a triplicated digital control system instead of an analog system as was originally provided in current BWR designs (BWR/2-6). The SB&PC system controls TCVs and turbine bypass valves to maintain reactor pressure. As presented in Subsection 15.2.4.2, no credible single failure in the control system results in a maximum demand to all actuators for all TCVs and bypass valves. A voter or actuator failure may result in an inadvertent opening of one TCV or one turbine bypass valve.

15.2.5.1.2 Sequence of Events and Systems Operation

The SB&PC system senses the pressure change and commands the remaining control valves to close, and thereby automatically mitigate the transient and maintain reactor power and pressure.

Table 15.2-20 lists the sequence of events for Figure 15.2-14

15.2.5.1.3 Core and System Performance

Reactor power and pressure is maintained. Reactor scram does not occur.

15.2.5.1.4 Barrier Performance

The effects of this event do not result in any temperature or pressure transient in excess of the criteria for which fuel, pressure vessel or containment are designed. The peak absolute pressure in the bottom of the vessel remains below the ASME code upset limit. Peak steam line absolute pressure near the SRVs remains below the setpoint of the SRVs.

15.2.5.1.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.5.2 Loss of Non-Emergency AC Power to Station Auxiliaries

This event bounds the Loss of Unit Auxiliary Transformer and Loss of Grid Connection events.

15.2.5.2.1 Identification of Causes

Causes for interruption or loss of power from the unit auxiliary transformer can arise from transformer (main and unit auxiliary) malfunction and isolated phase bus failures. Loss of grid connection can result from major shifts in electrical loads, loss of loads, lightning, storms, wind, etc., which contribute to electrical grid instabilities. These instabilities could cause equipment damage if unchecked. Protective relay schemes automatically disconnect electrical sources and loads to mitigate damage and regain electrical grid stability. The plant is designed to effect bus transfers and operate isolated from the electrical grid without scram. However in this analysis, it is assumed that concurrent with a load rejection, there is a simultaneous loss of power on the four power generation busses; this will cause the feedwater and circulating pumps to be lost. The bypass valves will remain initially available. The loss of the power generation busses produces a scram signal. The loss of the circulating water pumps results in a loss of condenser vacuum over a period of time. As condenser vacuum drops the turbine trips, bypass valves close and the MSIVs close.

15.2.5.2.2 Sequence of Events and Systems Operation

Sequence of Events

For the Loss of Unit Auxiliary Power Transformer, Table 15.2-21 lists the sequence of events for Figure 15.2-15.

Identification of Operator Actions

The operator should maintain the reactor water level by use of the IC system and Control Rod Drive system and control reactor pressure using the ICS and RWCU/SDC system. Verify that the turbine and generator DC oil pumps are operating satisfactorily to prevent turbine bearing damage. Also verify proper switching and loading of the standby diesel generators.

The following is the sequence of operator actions expected during the course of the events when no immediate restart is assumed. The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;

- Check that diesel generators start and carry their assigned loads;
- Monitor reactor water level and pressure; verify that Control Rod Drive flow is controlling water level;
- Use IC system to control pressure;
- Check turbine for proper operation of all auxiliaries during coastdown;
- Put the mode switch in the STARTUP position before the reactor pressure decays below 6 MPa (870 psig);
- Secure the IC when both reactor pressure and level are under control;

Systems Operation

This event, unless otherwise stated, assumes and takes credit for normal functioning of plant instrumentation and controls, plant protection and reactor protection systems.

The reactor is subjected to a complex sequence of events when the plant loses all auxiliary power. Estimates of the responses of the various reactor systems provide the simulation sequence shown in Table 15.2-21.

15.2.5.2.3 Core and System Performance

Figure 15.2-15 shows graphically the simulated transient. The initial portion of the transient is similar to the load rejection transient. At 2s the loss of the power generation busses signal produces a Scram and activation of the ICs. At about 6 seconds the turbine bypass valves are assumed no longer available to bypass the steam to the main condenser. The MSIV closure is produced at 14s due to low condense vacuum signal. The CRD high pressure injection is initiated due to low water level (Level 2), but the HPCRD flow is delayed until diesel power is available (145 seconds). There is no significant increase in fuel temperature. The number of rods in boiling transition remains within the acceptance criterion for AOOs. Hence, fuel thermal margins are not threatened and the design basis is satisfied. This event will be reanalyzed for each specific core configuration.

15.2.5.2.4 Barrier Performance

Peak nuclear system absolute pressure at the vessel bottom remains below the ASME code upset limit. Peak absolute pressure in the main steamline remains below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool.

15.2.5.2.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.5.3 Loss of All Feedwater Flow

Identification of Causes

A loss of FW flow could occur from pump failures, operator errors, or reactor system variables such as a high vessel water level (Level 8) trip signal.

15.2.5.3.1 Sequence of Events and Systems Operation

Sequence of Events

Table 15.2-22 lists the sequence of events for Figure 15.2-16.

Identification of Operator Actions

The operator should ensure ICS actuation and CRD injection transfer to high pressure injection mode so that water inventory is maintained in the reactor vessel. The operator should also monitor reactor water level, the pressure control, and the TG auxiliaries during shutdown.

The following is the sequence of operator actions expected during the course of the event when no immediate restart is assumed. The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Monitor reactor water level and pressure; verify that CRD flow is controlling water level;
- Verify IC system initiation; use the IC system to control pressure;
- Monitor turbine coastdown and break vacuum before the loss of steam seals (check turbine auxiliaries for proper operation);
- When desired, the RWCU/SDC system can be put into service.

Systems Operation

Loss of FW flow results in a reduction of vessel inventory, causing the vessel water level to drop. The first corrective action is the loss of the power generation busses scram trip actuation. This scram trip function meets the single-failure criterion.

15.2.5.3.2 Core and System Performance

The results of this transient simulation are presented in Figure 15.2-16. The initial water level is assumed at the L4 level. Feedwater flow terminates, and the loss of the power generation busses scram signal is assumed (with activation of the ICs simultaneously). Subcooling decreases, causing a reduction in core power level and pressure. As the core power level is reduced, the turbine steam flow starts to drop off because of the action of the pressure regulator in attempting to maintain pressure. Water level continues to drop, and the vessel level (Level 3) scram trip setpoint is reached. Note that the reactor has been scrammed previously. The vessel water level continues to drop to Level 2. At that time, CRD high pressure injection and closure of all MSIVs are produced (with 30s delay). The number of rods in boiling transition remains within the acceptance criterion for AOOs because increases in the heat flux are not experienced. Consequently, this event does not need to be reanalyzed for specific core configurations.

15.2.5.3.3 Barrier Performance

The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed. Therefore, these barriers maintain their integrity and function as designed.

15.2.5.3.4 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.6 AOO Analysis Summary

The results of the system response analyses are presented in Table 15.2-5. Based on these results, the limiting AOO events have been identified. The limiting events that establish the CPR operating limit include:

- Limiting pressurization event: MSIV Isolation
- Limiting decrease in core coolant temperature event: Loss of feedwater heating with SCRRI actuation

For the core loading in Figure 4.3-1, the resulting initial core MCPR operating limit is 1.30**.** The operating limit based on the initial core design will be provided by the operating license holder.

15.2.7 COL Information

The following potentially limiting AOOs are should be evaluated for core design changes:

- Loss of Feedwater Heating
- Closure of One Turbine Control Valve
- Turbine Trip with Bypass
- Inadvertent Isolation Condenser Initiation
- Loss of Non-Emergency AC to Station Auxiliaries

15.2.8 References

None.

Table 15.2-1

Input Parameters And Initial Conditions Used In AOO Analyses

Parameter	Value
Thermal Power Level, MWt	4500
Steam Flow, kg/h (Mlbm/hr) Rated Value Analysis Value	 8757 (19.31) 8766 (19.33)
Feedwater Flow Rate Rated Value, kg/h (Mlbm/hr) Analysis Total Flow For All Pumps Runout, % of rated at 1065 psig (At rated dome pressure, 1025 psig).	 8736 (19.27) 155 (164)
Feedwater Temperature, °C (°F) Rated FW Heating Temperature Loss	 216 (420) 55.6 (100)
Vessel Dome Pressure, MPaG (psig)	7.07 (1025)
Vessel Core Pressure, MPaG (psig)	7.17 (1040)
Turbine Bypass Capacity, % of rated	110
TCV Closure Times, seconds (Bounding) Fast Closure Analysis Value Assumed Slow Closure Analysis Value	 0.08 2.5
TSV Closure Times, seconds	0.100
% of Rated Steam Flow That Can Pass Through 3 Turbine Control Valves	85 (Partial Arc)
Core Coolant Inlet Enthalpy, kJ/kg Rated Value Analysis Value	 1195 1193
Turbine Inlet Pressure, MPaG (psig)	6.57
Fuel Lattice	N

Table 15.2-1

Input Parameters And Initial Conditions Used In AOO Analyses

Parameter	Value
Core Leakage Flow, %	9.4
MCPR Operating Limit	1.30
Control Rod Drive Position versus Time	Table 15.2-2 & 3
Nuclear Characteristics Used in TRACG Simulations	Middle of Cycle and End of Cycle
Safety/Relief Valve (SRV) capacity, %NBR At design pressure, MPaG (psig) Quantity Installed	89.5 8.618 (1250) 18
Safety Function Delay, seconds	0.2
Safety Function Opening Time, seconds	1.5
Analysis values for SRV setpoints Low Setpoint, MPaG (psig) High Setpoint, MPaG (psig)	 8.618 (1250) 8.756 (1270)
Closure Scram Position of 2 or More MSIVs, % open	85
MSIV Closure Times, seconds	3.0-5.0
High Flux Trip, % NBR	125.0
High Pressure Scram, MPaG (psig)	7.619 (1105)
Vessel level Trips (above bottom vessel) Level 9—(L9), m (in) Level 8—(L8), m (in) Level 4—(L4), m (in) Level 3—(L3), m (in) Level 2—(L2), m (in) Level 1.5—(L1.5), m (in) Level 1—(L1), m (in) Level 0.5 – (L0.5) m (in)	 22.39 (881.5) 21.89 (861.8) 20.60 (811.2) 19.78 (778.7) 16.05 (631.9) 13.00 (511.8) 10.00 (393.7) 8.45 (332.7)

Table 15.2-1

Input Parameters And Initial Conditions Used In AOO Analyses

Parameter	Value
APRM Simulated Thermal Power Trip Scram, % NBR Time Constant, s	 115 7
Total Steamline Volume, m^3 (ft^3)	135 (4767)
Isolation Condensers Analysis Temperature, °C (°F) Analysis Enthalpy, kJ/kg (Btu/lbm) Analysis Time To Reach Full Loop Flow, s Heat Removal Capacity, % Rated Power	 40 (104) 174 (75) 31 (1) 3

(1) In the analysis a 15s delay has been included to ensure the ICs do not remove heat before 15s after activation as indicated in the requirements of ICs. For Loss of Feedwater and Loss Of Off-site Power this delay has been reduced to 5s to better model the cool water drop in the vessel from the ICs at the beginning of the event.

Table 15.2-2

CRD Scram Times for Vessel Bottom Pressures Below 7.481 MPa gauge (1085 psig)

Rod Insertion (%)	**Scram Time (seconds) (After De-energization (0.05 s)) Used in Analysis**
0	0.0
10	0.34
40	0.80
60	1.15
100	2.23

Table 15.2-3

CRD Scram Times for Bottom Vessel Pressures Between 7.481 MPa gauge (1085 psig) and 8.618 MPa gauge (1250 psig)

Rod Insertion (%)	**Scram Time (seconds) (After De-energization (0.05 s)) Used in Analysis**
0	0.0
10	0.37
40	0.96
60	1.36
100	2.95

Table 15.2-4

Sequence of Events for Loss of Feedwater Heating

Time (s)	Event
0	Initiate a 55.6°C (16.7°C) temperature reduction in the FW system.
22 (est)	RC&IS initiates Selected Control Rod Run-In
25 (est.)	Initial effect of unheated FW starts to raise core power level.
23.0 57.0 89.0	SCRRI group start insertion
123.0 157.0 189.0	SCRRI group finish insertion
300.0 (est.)	Reactor variables settle into new steady state.

* See Figure 15.2-1

Table 15.2-5

Results Summary of Anticipated Operational Occurrence Events

Sub-section I.D.	Description	Max. Neutron Flux, % NBR	Max. Dome Pressure, MPaG (psig)	Max. Vessel Bottom Pressure, MPaG (psig)	Max. Steamline Pressure, MPaG (psig)	Max. Core Average Surface Heat Flux, % of Initial	ΔCPR
15.2.1.1	Loss of Feedwater Heating	114	7.09 (1028)	7.23 (1049)	7.05 (1023)	114	0.10
15.2.2.1	Closure of One Turbine Control Valve. FAST/SLOW	121 109	7.20 (1044) 7.20 (1044)	7.34 (1065) 7.34 (1065)	7.16 (1038) 7.16 (1038)	102 102	0.04 0.03
15.2.2.2	Generator Load Rejection with Turbine Bypass	114	7.09 (1028)	7.22 (1048)	7.14 (1036)	113	0.04
15.2.2.3	Generator Load Rejection with a Single Failure in the Turbine Bypass System	150	7.39 (1072)	7.52 (1091)	7.38 (1070)	102	0.03
15.2.2.4	Turbine Trip with Bypass	114	7.08 (1027)	7.21 (1046)	7.07 (1026)	113	0.04
15.2.2.5	Turbine Trip with a Single Failure in the Turbine Bypass System	126	7.36 (1067)	7.49 (1086)	7.35 (1066)	101	0.02
15.2.2.6	Closure of One MSIV	112	7.16 (1038)	7.30 (1059)	7.12 (1033)	101	0.03
15.2.2.7	Closure of All MSIV	104	7.83 (1136)	7.96 (1155)	7.83 (1136)	100	≤ 0.01
15.2.2.8	Loss of Condenser Vacuum	106	7.09 (1028)	7.23 (1049)	7.07 (1025)	100	≤ 0.01
15.2.2.9	Loss of Shutdown Cooling Function of RWCU/SDC	{ - - }	{ - - }	{ - - }	{ - - }	{ - - }	{ - - }
15.2.4.1	Inadvertent Isolation Condenser Initiation	110	7.08 (1027)	7.22 (1047)	7.04 (1021)	107	0.08
15.2.4.2	Runout of One Feedwater Pump	103	7.08 (1027)	7.22 (1047)	7.05 (1023)	100	0.02
15.2.5.1	Opening of One Turbine Control or Bypass Valve	105	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	≤ 0.01
15.2.5.2	Loss of Non-Emergency AC Power to Station Auxiliaries	122	7.11 (1032)	7.25 (1052)	7.24 (1046)	100	0.03
15.2.5.3	Loss of Feedwater Flow	100	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	≤ 0.01

Table 15.2-6

Sequence of Events for Fast Closure of One Turbine Control Valve

Time (sec)	**Event***
0	Simulate one main TCV to fast close.
0	Failed TCV starts to close.
0.10	TCV closed (Realistic closure timing)
1.05	Turbine bypass valves start to open.
10.0	New steady state is established

* See Figure 15.2-2.

Table 15.2-7

Sequence of Events for Slow Closure of One Turbine Control Valve

Time (sec)	**Event***
0	Simulate one main TCV to slow close.
0	Failed TCV starts to close.
2.5	TCV closed
2.82	Turbine bypass valves start to open.
10.0	New steady state is established.

* See Figure 15.2-3.

Table 15.2-8

Sequence of Events for Generator Load Rejection with Turbine Bypass

Time (sec)	**Event***
-0.015	Turbine-generator detection of loss of electrical load.
0.0	Turbine-generator load rejection sensing devices trip to initiate TCVs fast closure and main turbine bypass system operation.
0.02	Turbine bypass valves start to open.
0.08	Turbine control valves closed.
0.22	SCRRI activated
1. 35. 68.	SCRRI groups start insertion
60	FW temperature is decreasing because of loss of turbine extraction steam to FW heaters
101. 135. 168.	SCRRI groups finish insertion
300	New steady state is established

* See Figure 15.2-4.

Table 15.2-9

Sequence of Events for Generator Load Rejection with a Single Failure in the Turbine Bypass System

Time (sec)	**Event***
-0.015	Turbine-generator detection of loss of electrical load.
0.0	Turbine-generator load rejection sensing devices trip to initiate TCVs fast closure and main turbine bypass system operation.
0.02	Turbine bypass valves start to open (Half fail to open).
0.08	Turbine control valves closed.
0.22	Not enough turbine bypass availability is detected and the plant is scrammed
0.40	Control Rods begin to enter in the core.
Long term	L2 is reached and HPCRD is activated to recover the level

* See Figure 15.2-5.

Table 15.2-10

Sequence of Events for Turbine Trip with Turbine Bypass

Time (sec)	**Event** *
0.0	Turbine trip initiates closure of main stop valves.
0.0	Turbine trip initiates bypass operation.
0.02	Turbine bypass valves start to open to regulate pressure.
0.1	Turbine stop valves closed.
0.22	SCRRI activated
1. 35. 68.	SCRRI groups start insertion
60	FW temperature is decreasing because of loss of turbine extraction steam to Fw heaters
101. 135. 168.	SCRRI groups finish insertion
300	New steady state is established

* See Figure 15.2-6.

Table 15.2-11

Sequence of Events for Turbine Trip with a Single Failure in the Turbine Bypass System

Time (sec)	**Event** *
0.0	Turbine trip initiates closure of main stop valves.
0.0	Turbine trip initiates bypass operation.
0.02	Turbine bypass valves start to open to regulate pressure (Half fail to open).
0.1	Turbine stop valves closed.
0.22	Not enough turbine bypass availability is detected and the plant is scrammed
0.40	Control Rods begin to enter in the core.
Long term	L2 is reached and HPCRD is activated to recover the level

* See Figure 15.2-7.

Table 15.2-12

Sequence of Events for Closure of one MSIV

Time (sec)	**Event** *
0.0	Closure of one MSIV.
2.0	Maximum neutron flux
2.9	Turbine Bypass open.
3.0	MSIV is closed
40.0	New steady state is reached

* See Figure 15.2-8.

Table 15.2-13

Sequence of Events for Closure of all MSIV

Time (sec)	**Event** *
0.0	Closure of all MSIVs (MSIV).
0.78	MSIVs reach 85% open.
0.85	MSIVs position trip scram initiated.
3.0	MSIVs are closed
4.80	Reactor pressure reaches its peak value.
31.82	The ICs are at full operation .
Long term	L2 activates HPCRD to recover the water level

* See Figure 15.2-9.

Table 15.2-14

Typical Rates of Decay for Loss of Condenser Vacuum

Cause	Estimated Vacuum Decay Rate
Failure of Isolation of Steam Jet Air Ejectors	1 inch/minute
Loss of One or More Circulating Water Pumps	0.5 inch/sec

Table 15.2-15

Sequence of Events for Loss of Condenser Vacuum

Time (sec)	**Event ***
-3.0	Initiate simulated loss of condenser vacuum trip.
0.0	Low condenser vacuum main turbine trip and Scram actuated.
0.02	Turbine bypass valves start to open to regulate pressure.
0.1	Turbine stop valves close.
0.22	Scram initiated
6.0	Low condenser vacuum initiates main turbine bypass valve closure.
6.5	Bypass valve is closed
8.0	Low condenser vacuum initiates MSIV closure.
11.60	L2 water level is reached.
21.81	HPCRD is activated
39.78	The ICs are at full operation
Long term	L2 activates HPCRD to recover the level

* See Figure 15.2-10.

Table 15.2-16

Trip Signals Associated With Loss of Condenser Vacuum

Vacuum (cm/in. of Hg)	**Protective Action Initiated**
5.2-9.1/2-3.6	Normal Vacuum Range
17.8-25.4/7-10	Main Turbine Trip (Stop Valve Closure) & Scram
50.8-58.4/20-23	MSIV Closure and Turbine Bypass Valve Closure

Table 15.2-17

Sequence of Events for Inadvertent Isolation Condenser Initiation

Time (sec)	**Event ***
0	Simulate IC cold water injection.
31	Full flow established for IC.
150	Power increase effect stabilized.

* See Figure 15.2-11.

Table 15.2-18

Single Failure Modes for Digital Controls

Modes	Description	Effects
1	Critical input failure	None—Signal from redundant transmitter is utilized—Operator informed of failure
2	Input failure while one sensor out of service	Possible system failure. Adverse effects minimized when possible
3	Operator switch single contact failure	None—Triplicated contacts
4	Processor channel failure	None—Redundant processors maintain control; Operator informed of failure
5	Processor failure while one channel out of service	System failure
6	Voter failure	Loss of control of one actuator (i.e., one FW pump only). FWCS locks up actuators.
7	Actuator failure	Loss of one actuator (i.e., one FW pump only)

Table 15.2-19

Sequence of Events for Runout of One Feedwater Pump

Time (sec)	**Events** *
0	Initiate simulated runout of one FW pump (at system design pressure) the pump runout flow is 75% of rated FW flow).
~0.1	Feedwater controller starts to reduce the FW flow from the FW pumps.
6.0	Vessel water level reaches its peak value and starts to return to its normal value.
21.0 (est.)	Vessel water level returns to its normal value.

* See Figure 15.2-13.

Table 15.2-20

Sequence of Events for Opening of one Turbine Control or Bypass Valve

Time (sec)	**Events** *
0	One Turbine Bypass opens
~0.1	TCV closes slightly to control pressure
30.0	New steady state is established

* See Figure 15.2-14.

Table 15.2-21
Sequence of Events for Loss of Non-Emergency AC Power to Station Auxiliaries

Time (sec)	**Event** *
0.0	Loss of AC power to station auxiliaries, which initiates a generator trip.
0.0	Additional Failure assumed in transfer to "Island mode", Feedwater, condensate and circulating water pumps are tripped.
0.0	Turbine control valve fast closure is initiated.
0.0	Turbine control valve fast closure initiates main turbine bypass system operation.
0.0	Feedwater and condenser pumps are tripped.
0.02	Turbine bypass valves start to open.
0.08	Turbine control valves closed.
2.0	Loss of power on the four power generation busses is detected and initiates a reactor scram and activation of ICs.
5.0	Feedwater flow decay to 0.
6.0	Low condenser vacuum setpoint is detected and initiates turbine bypass closure.
6.0	Loss of condenser Vacuum rate is reduced due to bypass valve closure
6.61	Vessel water level reaches Level 3
7.0	ICs drops cold water inside the vessel
9.9	Vessel water level reaches Level 2.
14.0	Low-Low condenser vacuum signal closes the MSI valves .
33.0	ICs are at rated flow
145.0	CRD high pressure injection mode is initiated.
≈100	The level recovers above 13m
≈850	The level recovers above 15m

* See Figure 15.2-15. This Figure has 50s of steady state to change the initial water level to L4

Table 15.2-22

Sequence of Events for Loss of All Feedwater Flow

Time (sec)	**Event** *
0	Trip of all FW pumps initiated.
2.0	Non FW flow availability initiates reactor scram and initiates IC
5.0	Feedwater flow decays to zero.
9.68	Vessel water level reaches Level 2.
19.68	CRD high pressure injection mode is also initiated.
33.0	The ICs are at full operation
39.69	MSIV closure and ICs are initiated.
≈100	The level recovers above 13m
≈800	The level recovers above 15m

* See Figure 15.2-16. This Figure has 50s of steady state to change the initial water level to L4.

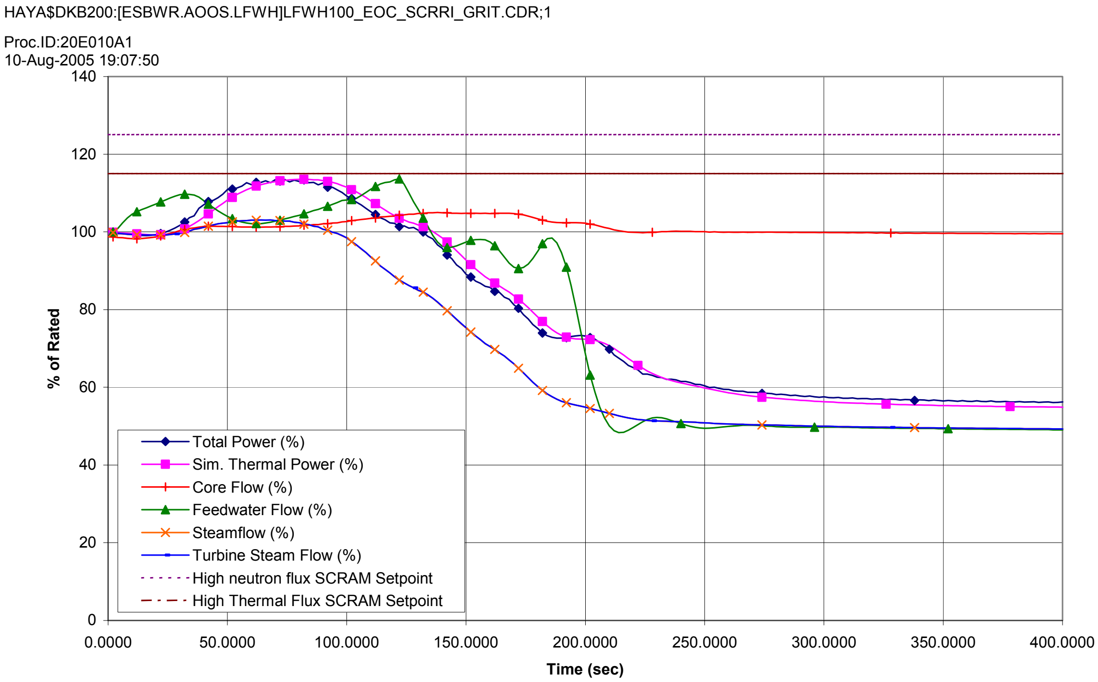


Figure 15.2-1a. Loss of Feedwater Heating

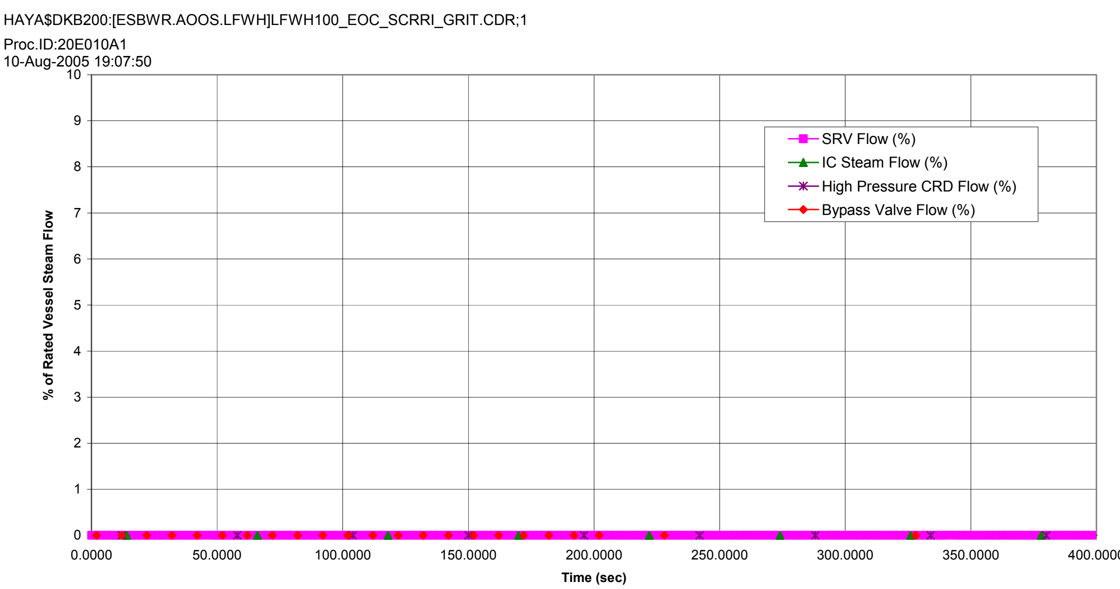


Figure 15.2-1b. Loss of Feedwater Heating

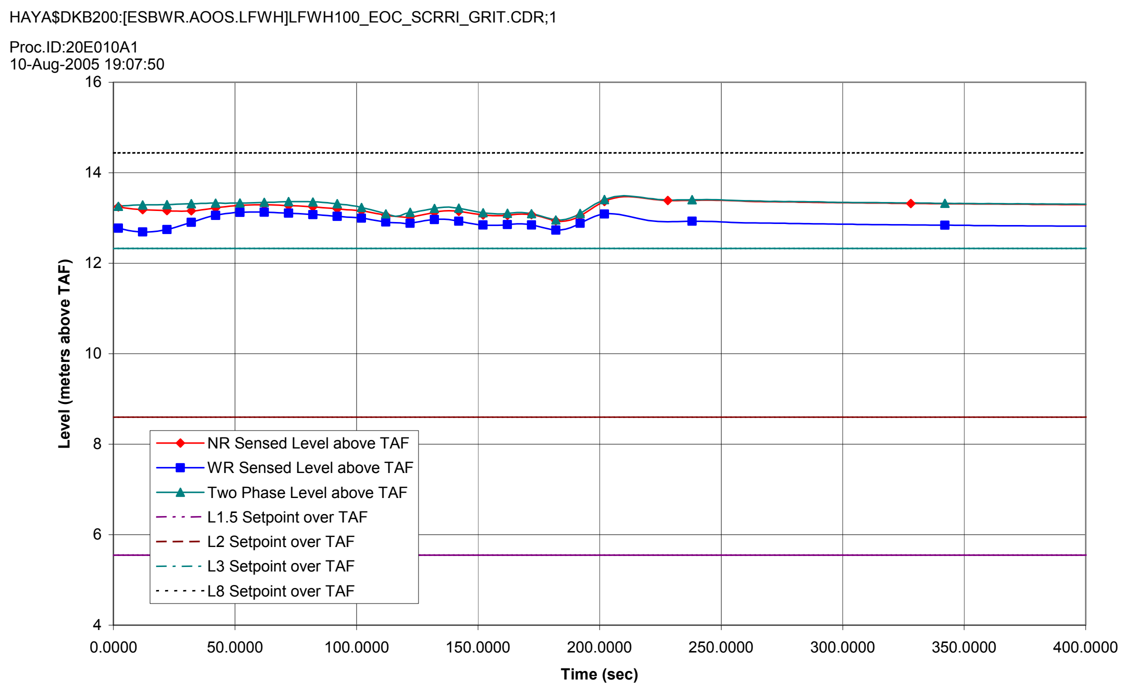


Figure 15.2-1c. Loss of Feedwater Heating

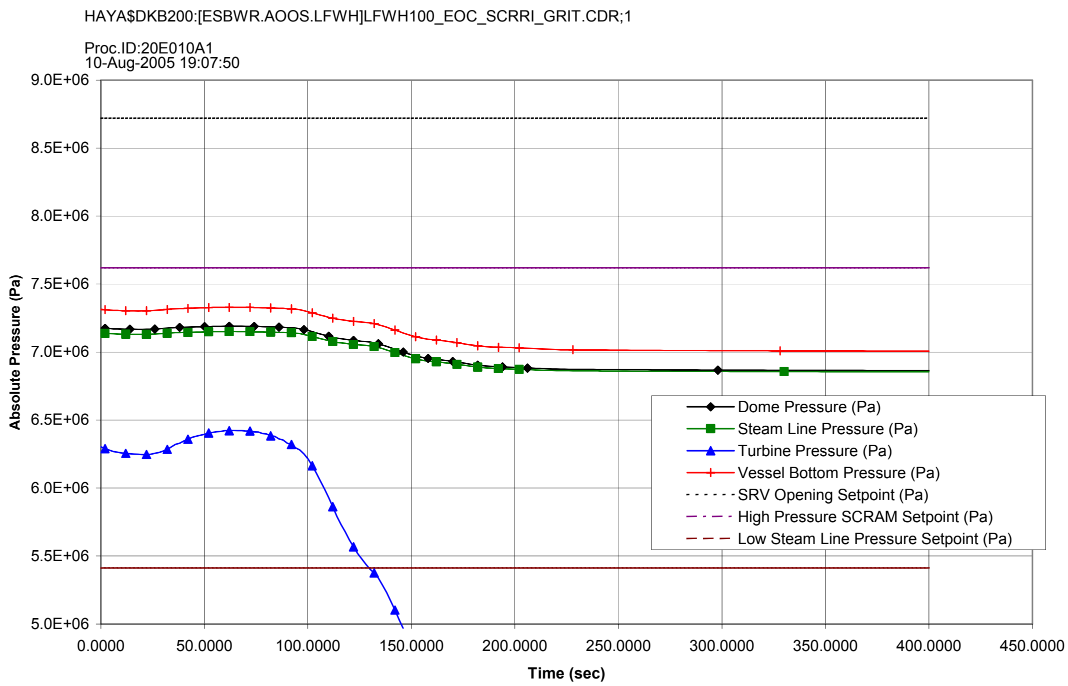


Figure 15.2-1d. Loss of Feedwater Heating

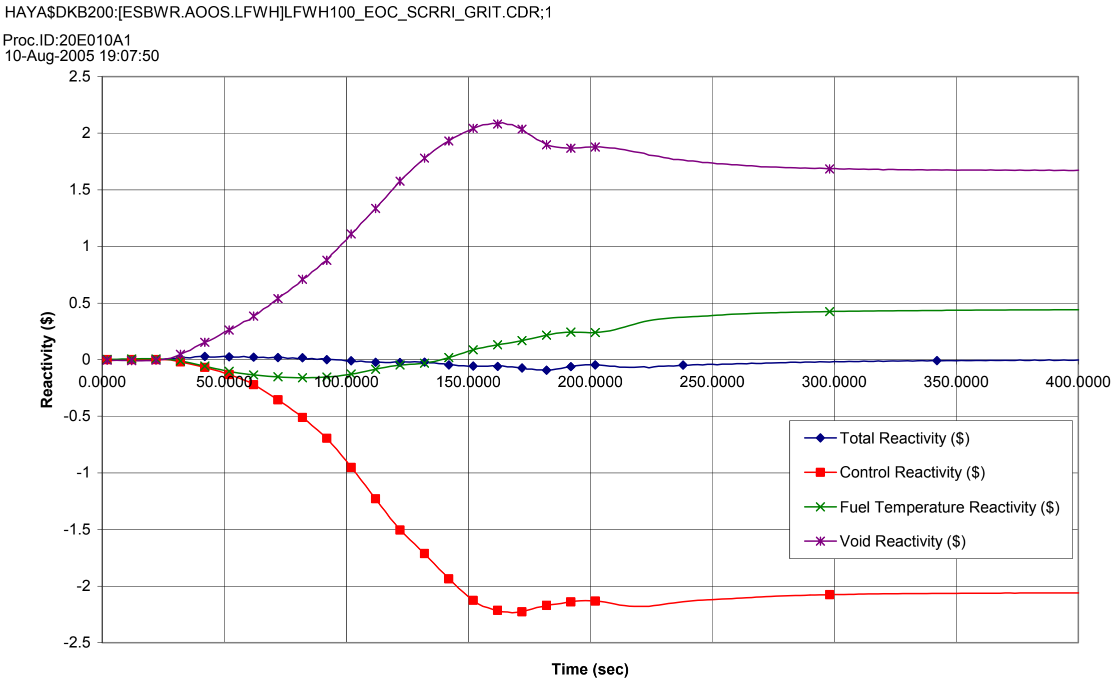


Figure 15.2-1e. Loss of Feedwater Heating

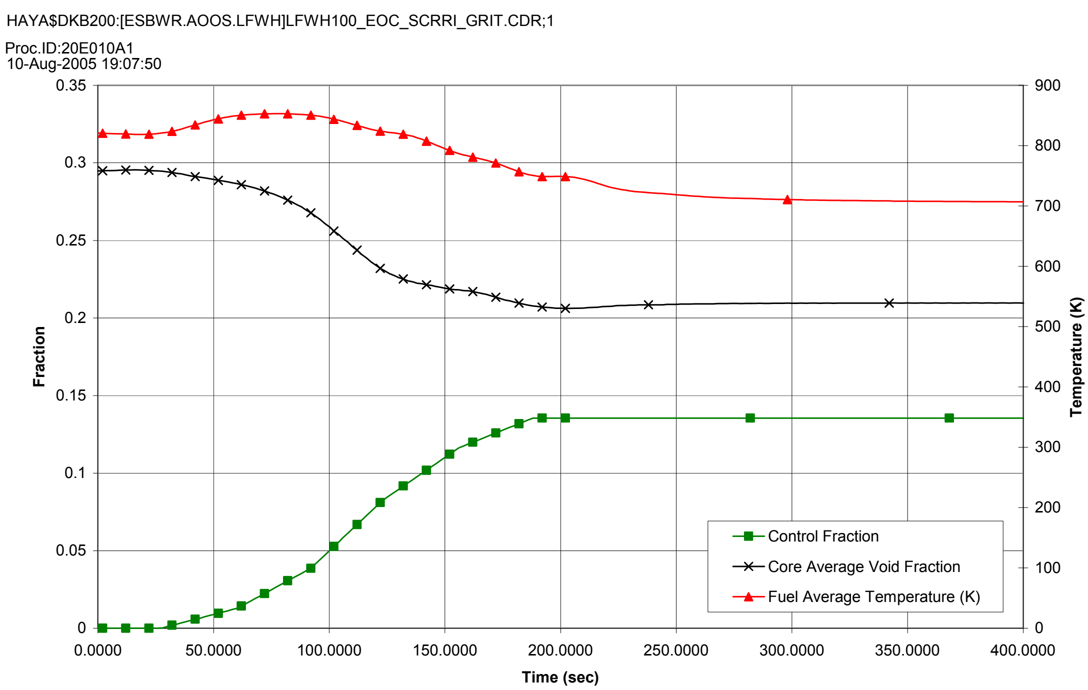


Figure 15.2-1f. Loss of Feedwater Heating

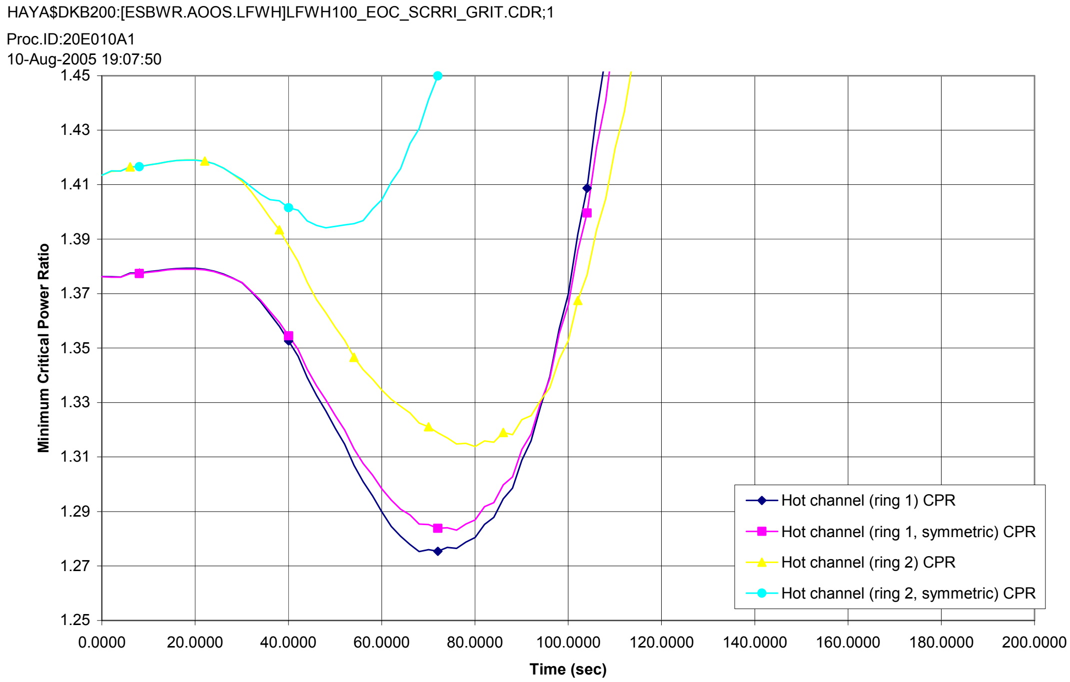


Figure 15.2-1g. Loss of Feedwater Heating

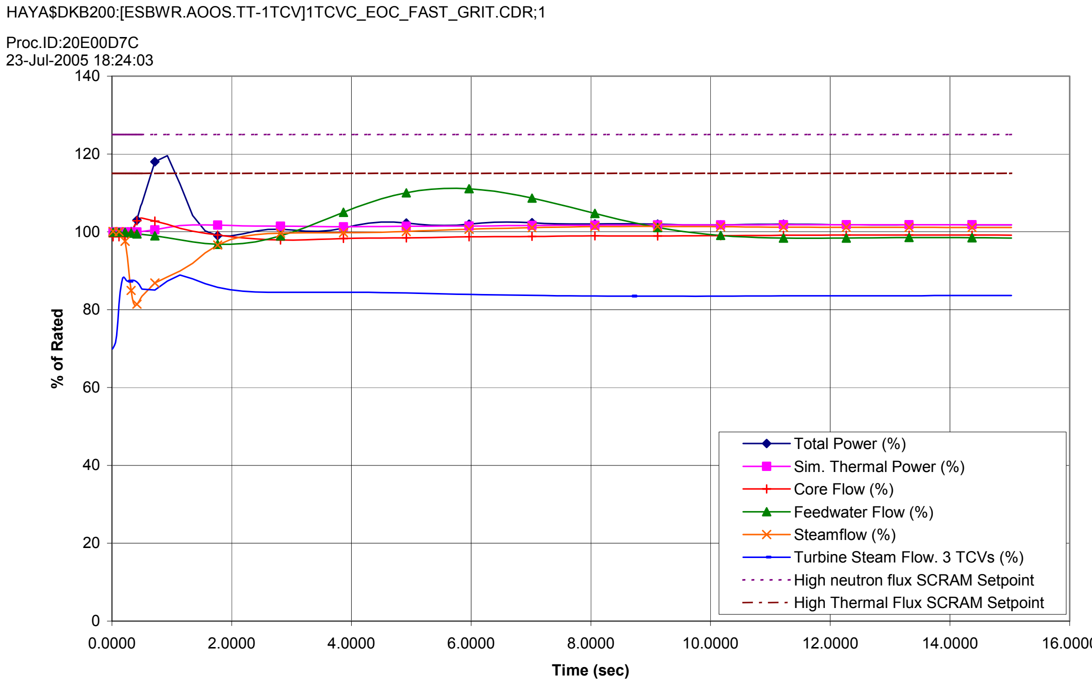


Figure 15.2-2a. Fast Closure of One Turbine Control Valve

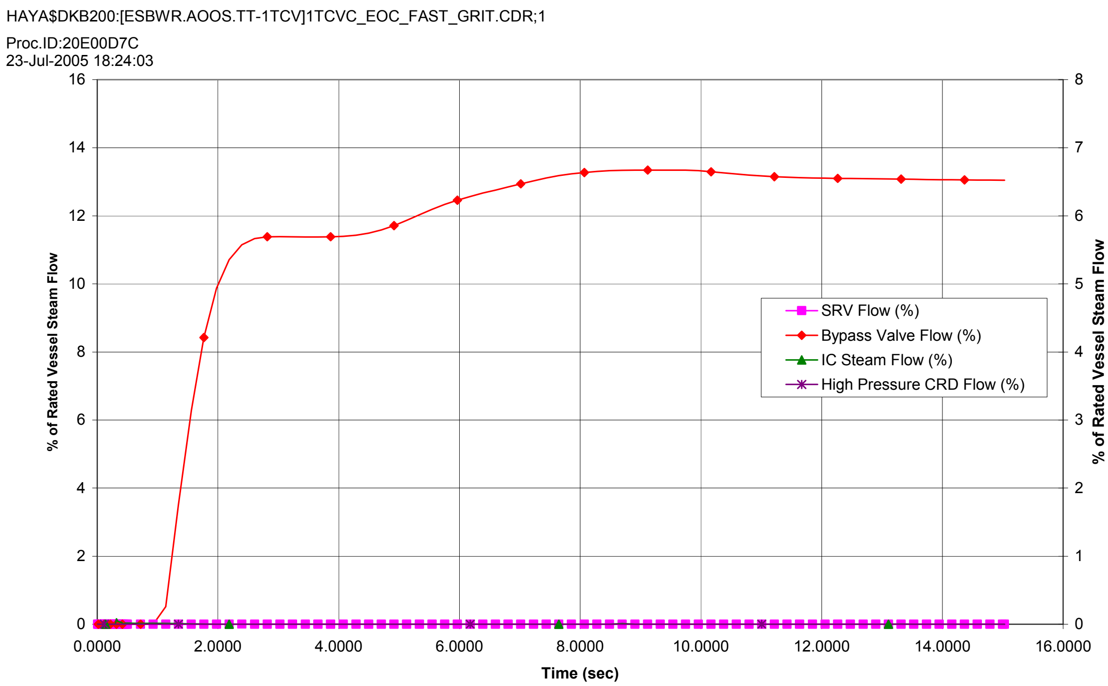


Figure 15.2-2b. Fast Closure of One Turbine Control Valve

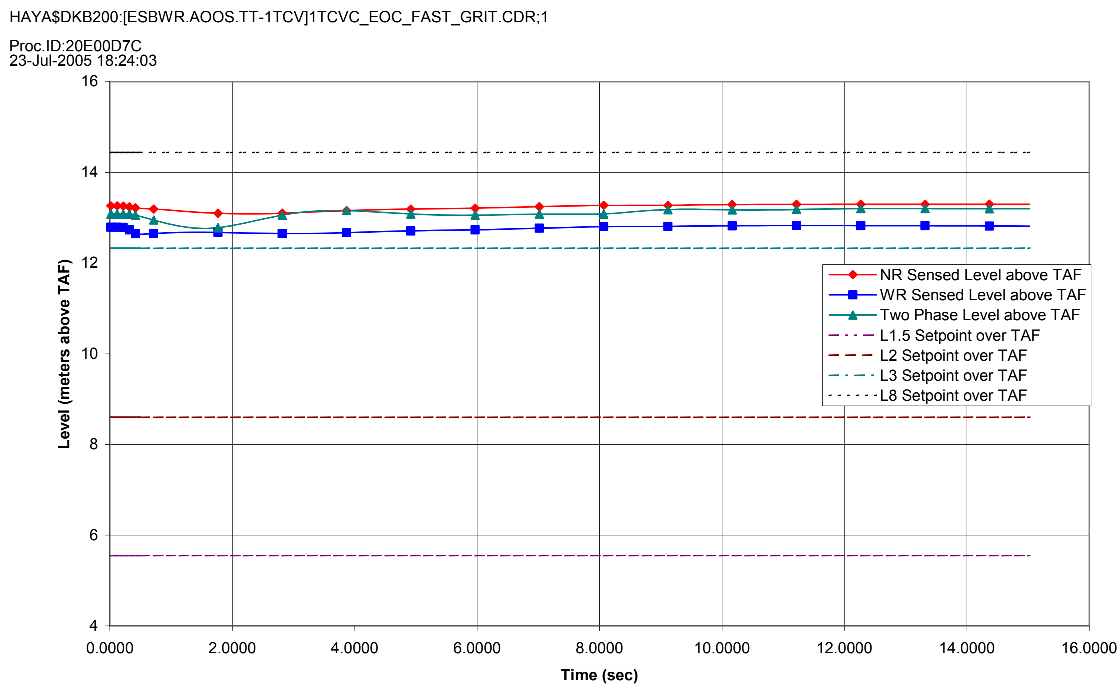


Figure 15.2-2c. Fast Closure of One Turbine Control Valve

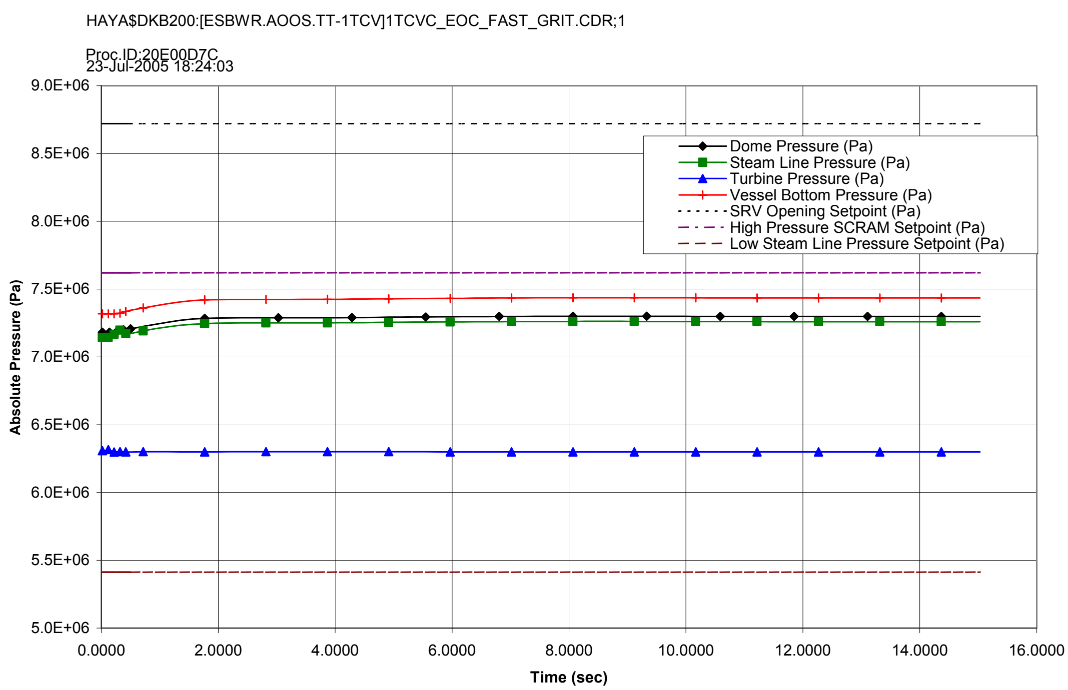


Figure 15.2-2d. Fast Closure of One Turbine Control Valve

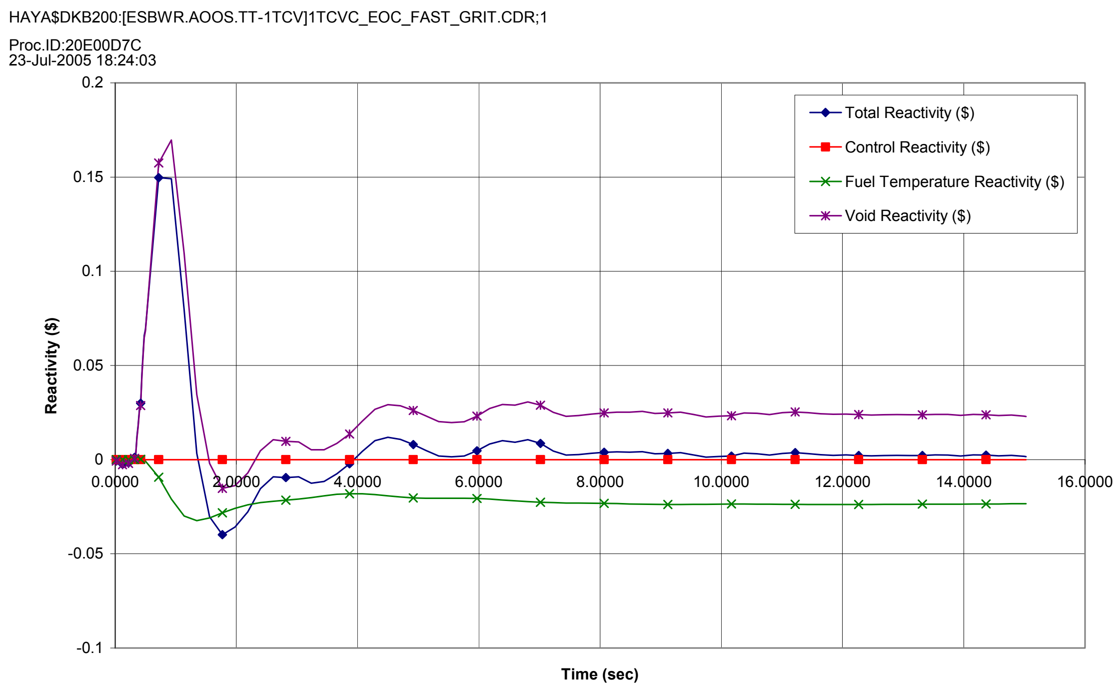


Figure 15.2-2e. Fast Closure of One Turbine Control Valve

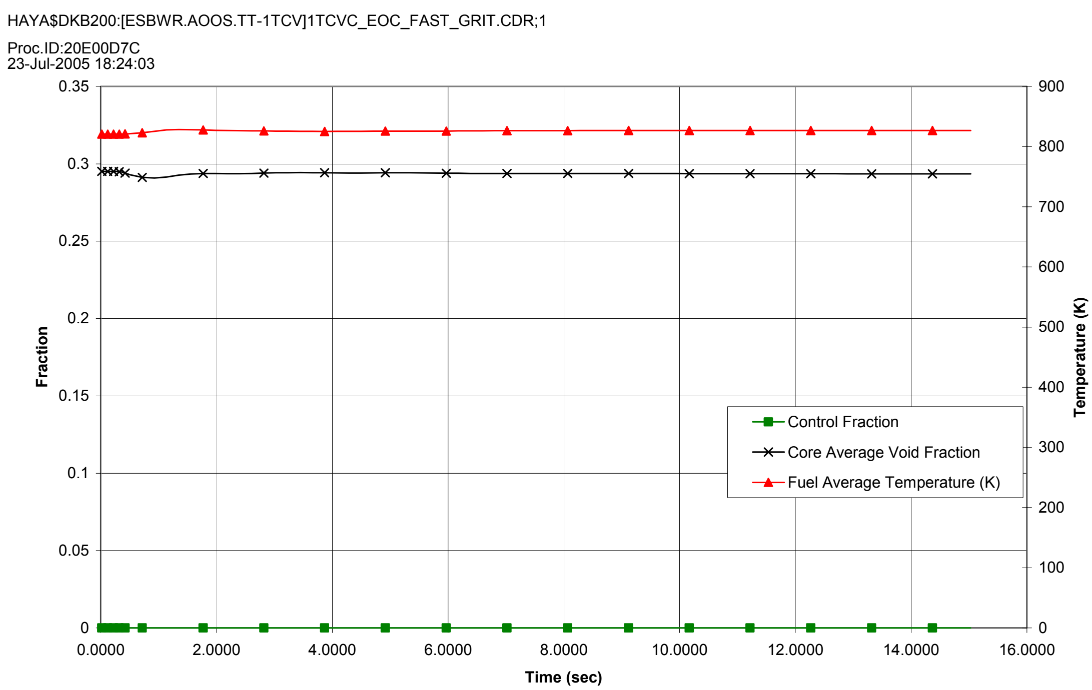


Figure 15.2-2f. Fast Closure of One Turbine Control Valve

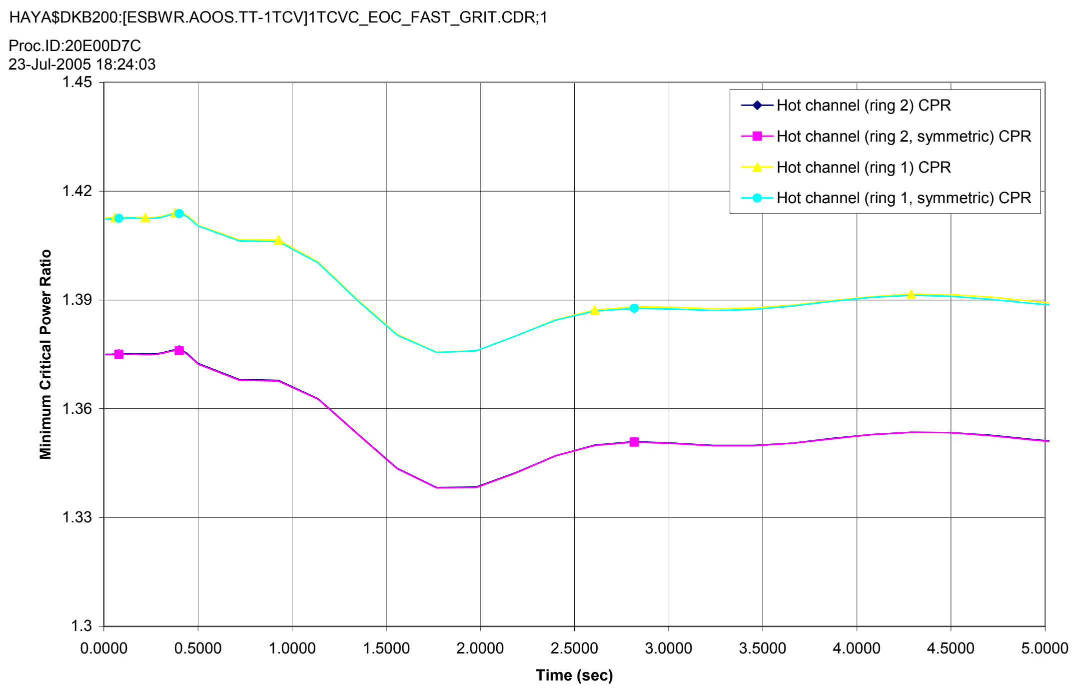


Figure 15.2-2g. Fast Closure of One Turbine Control Valve

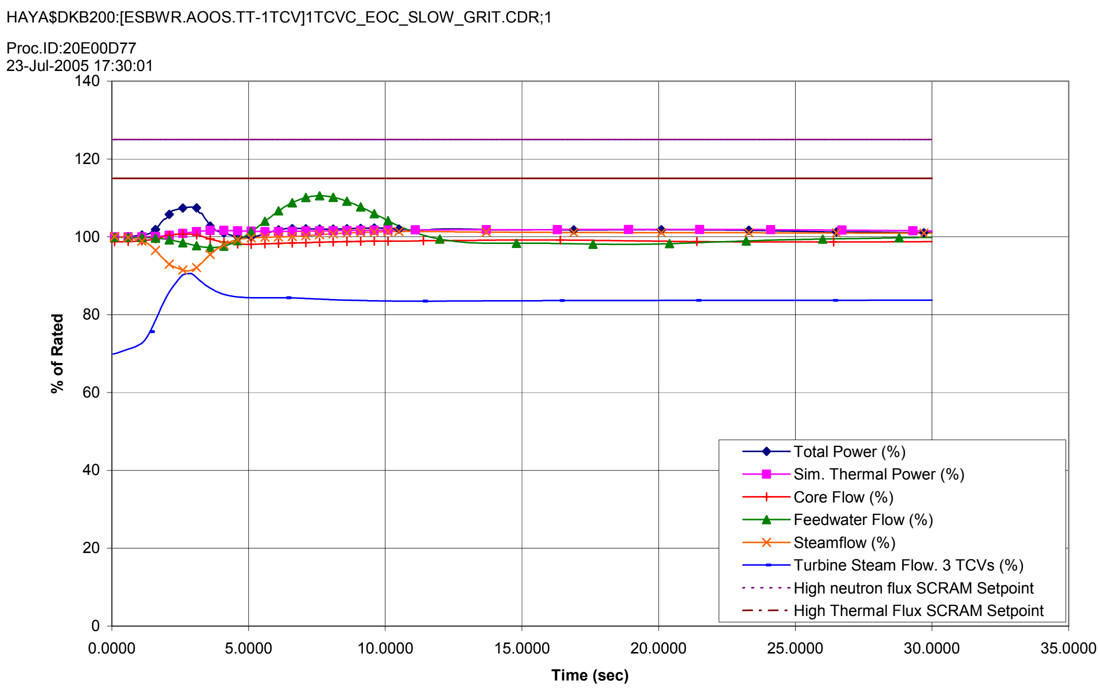


Figure 15.2-3a. Slow Closure of One Turbine Control Valve

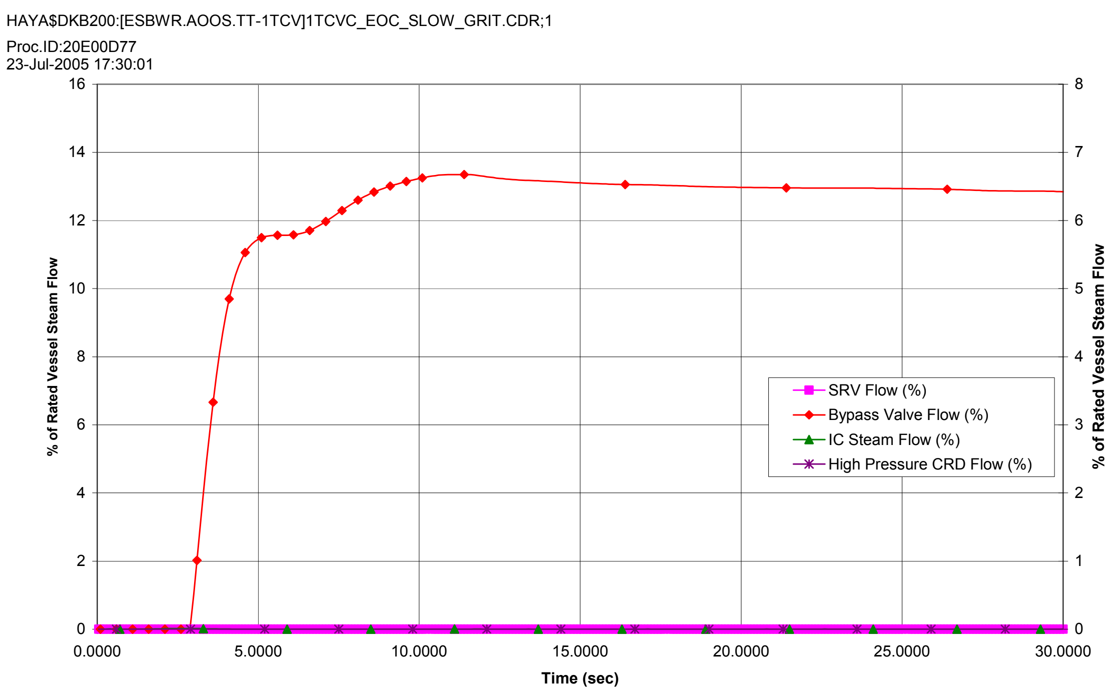


Figure 15.2-3b. Slow Closure of One Turbine Control Valve

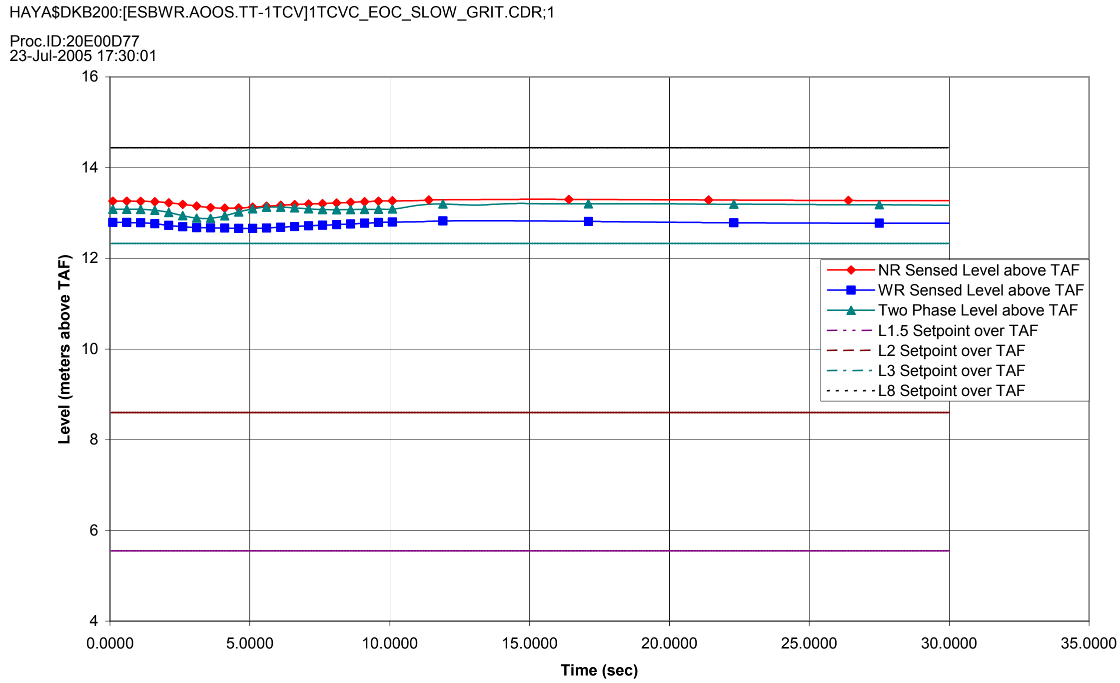


Figure 15.2-3c. Slow Closure of One Turbine Control Valve

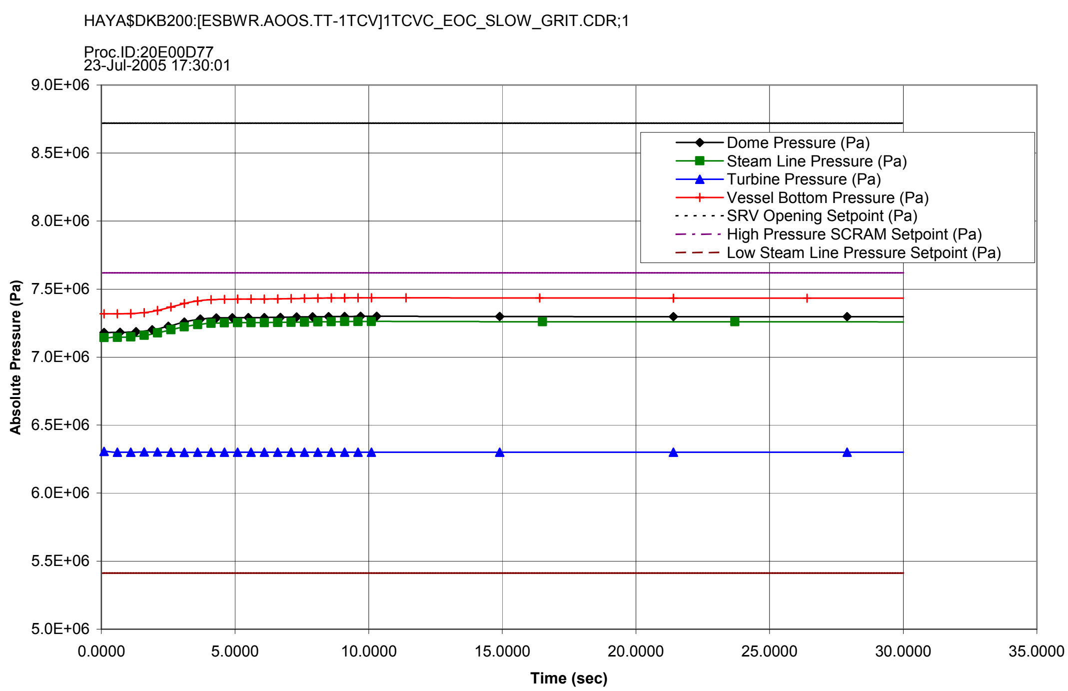


Figure 15.2-3d. Slow Closure of One Turbine Control Valve

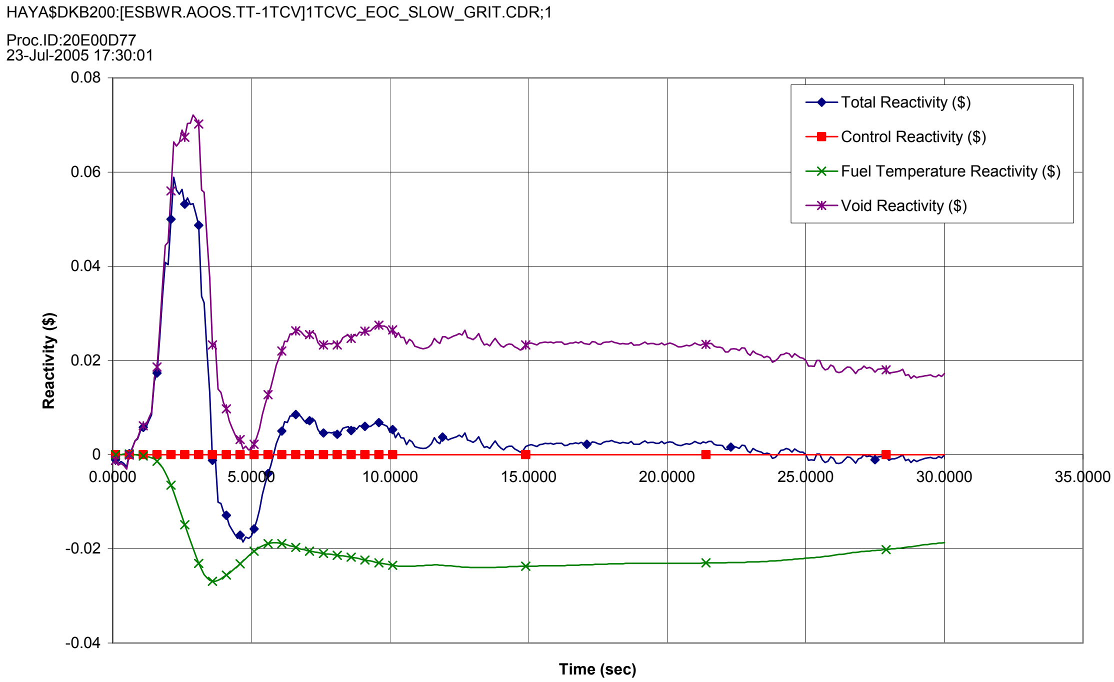


Figure 15.2-3e. Slow Closure of One Turbine Control Valve

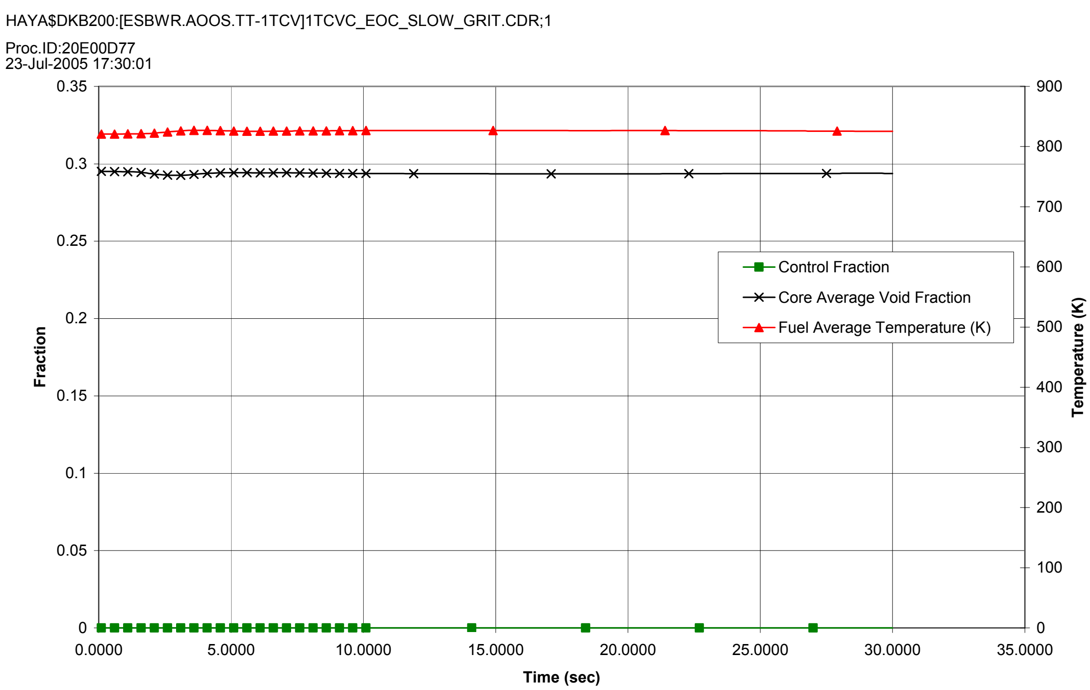


Figure 15.2-3f. Slow Closure of One Turbine Control Valve

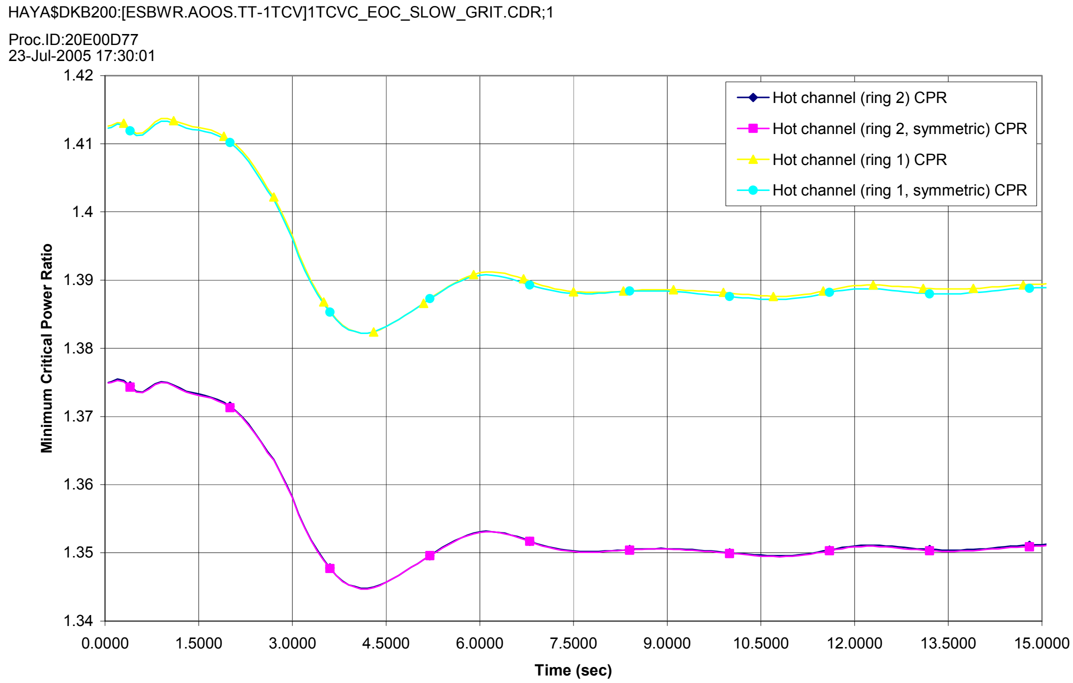


Figure 15.2-3g. Slow Closure of One Turbine Control Valve

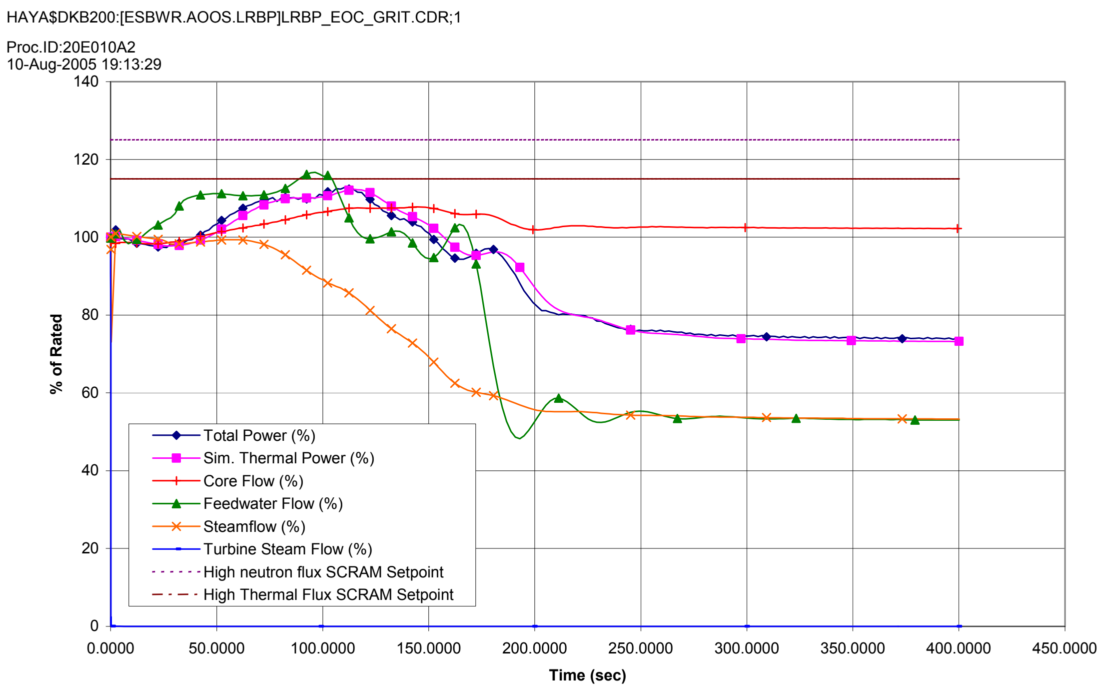


Figure 15.2-4a. Generator Load Rejection with Turbine Bypass

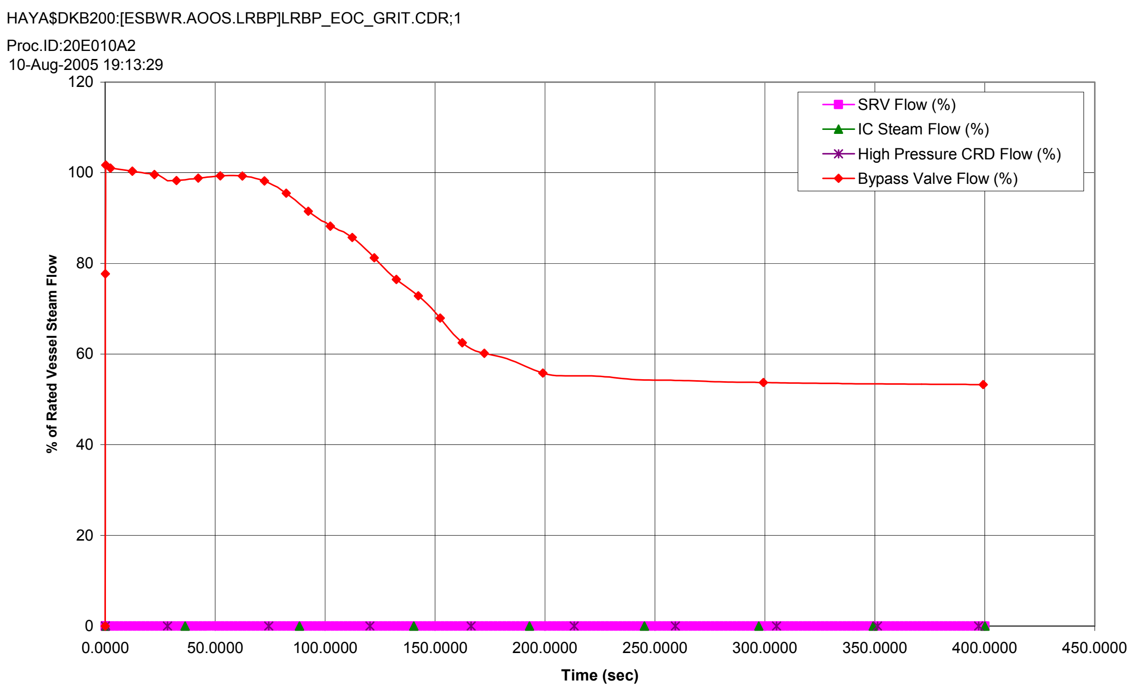


Figure 15.2-4b. Generator Load Rejection with Turbine Bypass

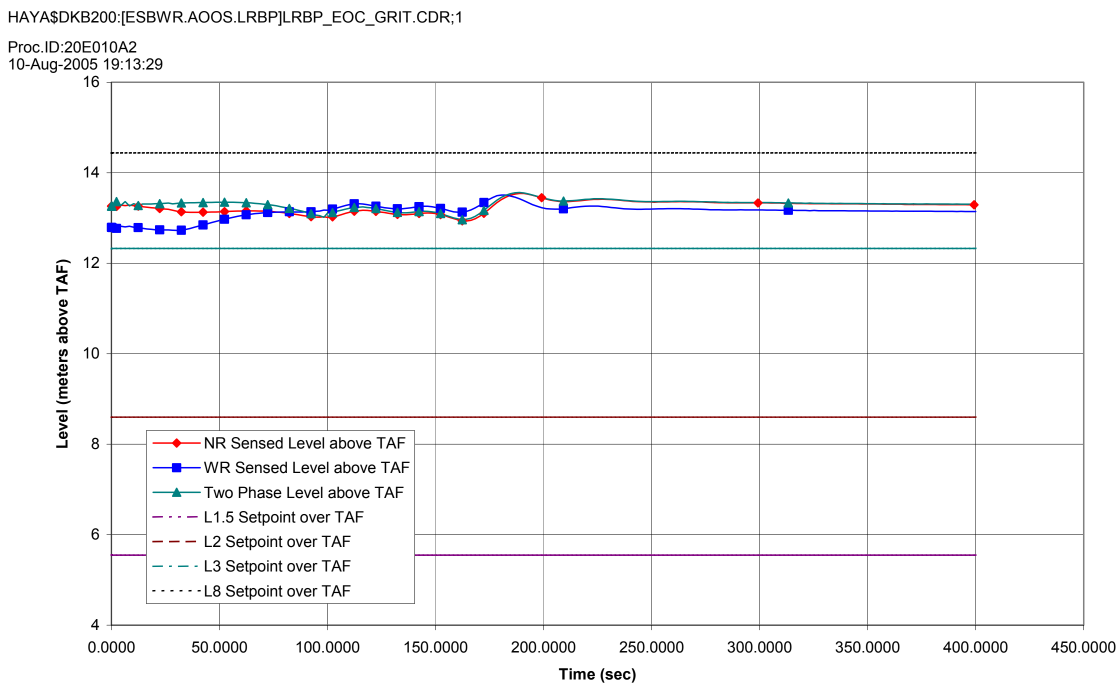


Figure 15.2-4c. Generator Load Rejection with Turbine Bypass

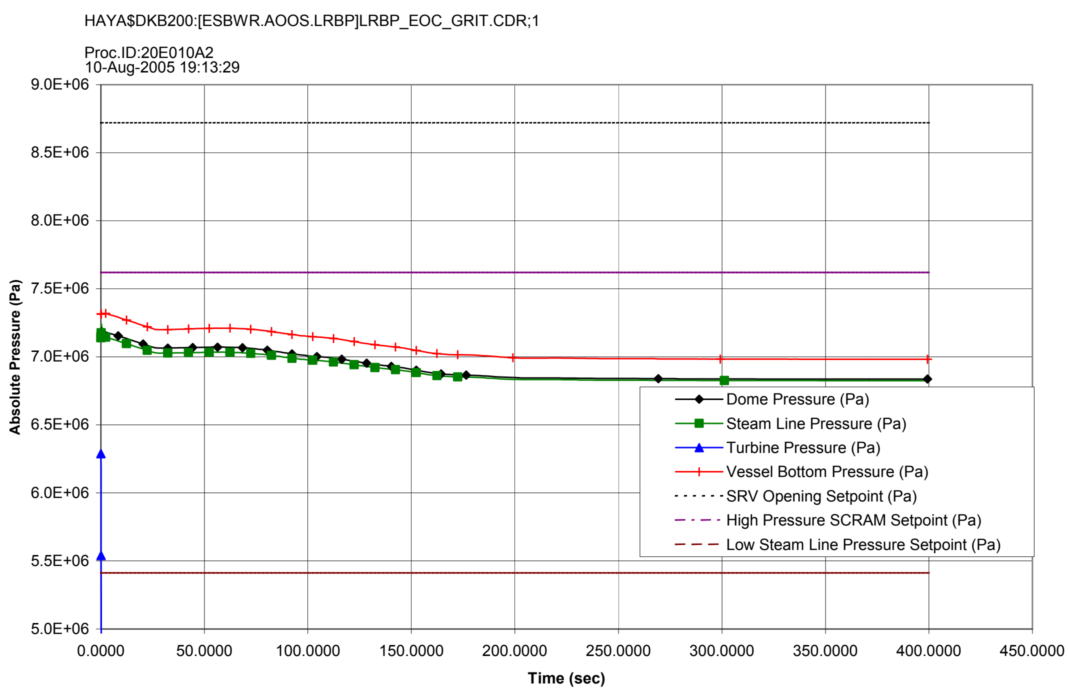


Figure 15.2-4d. Generator Load Rejection with Turbine Bypass

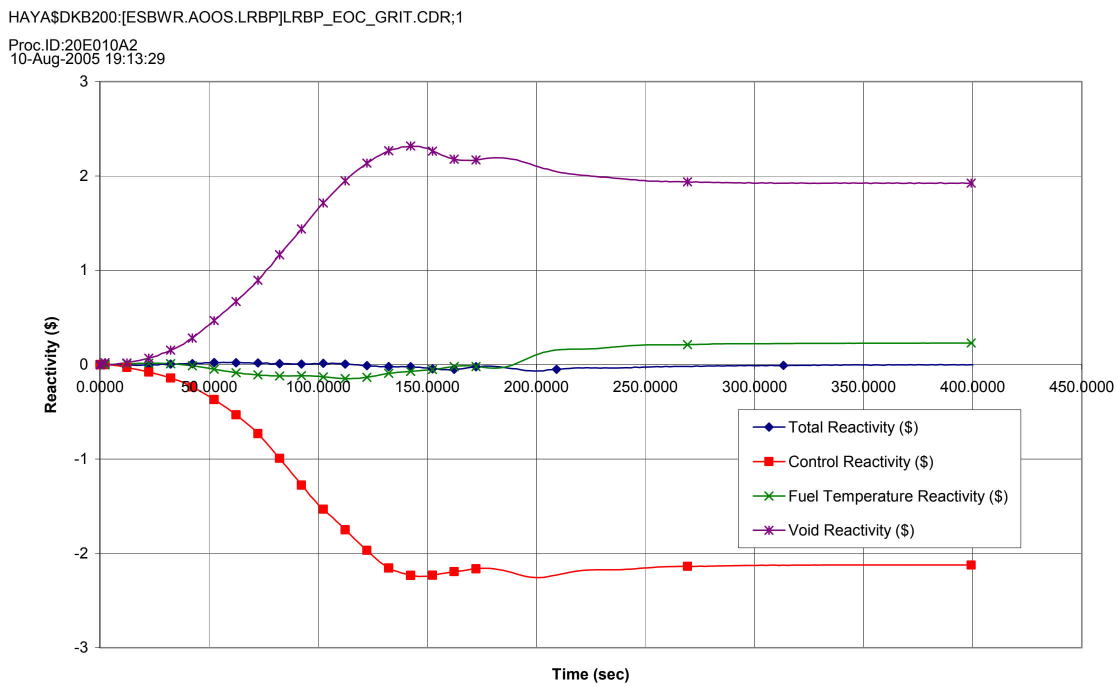


Figure 15.2-4e. Generator Load Rejection with Turbine Bypass

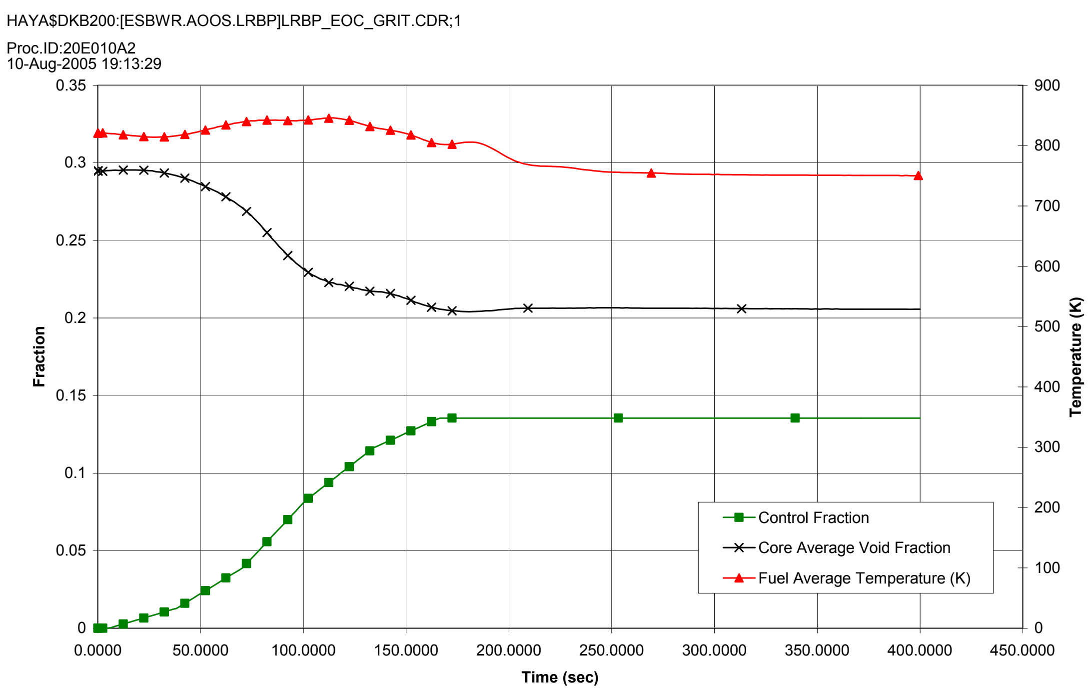


Figure 15.2-4f. Generator Load Rejection with Turbine Bypass

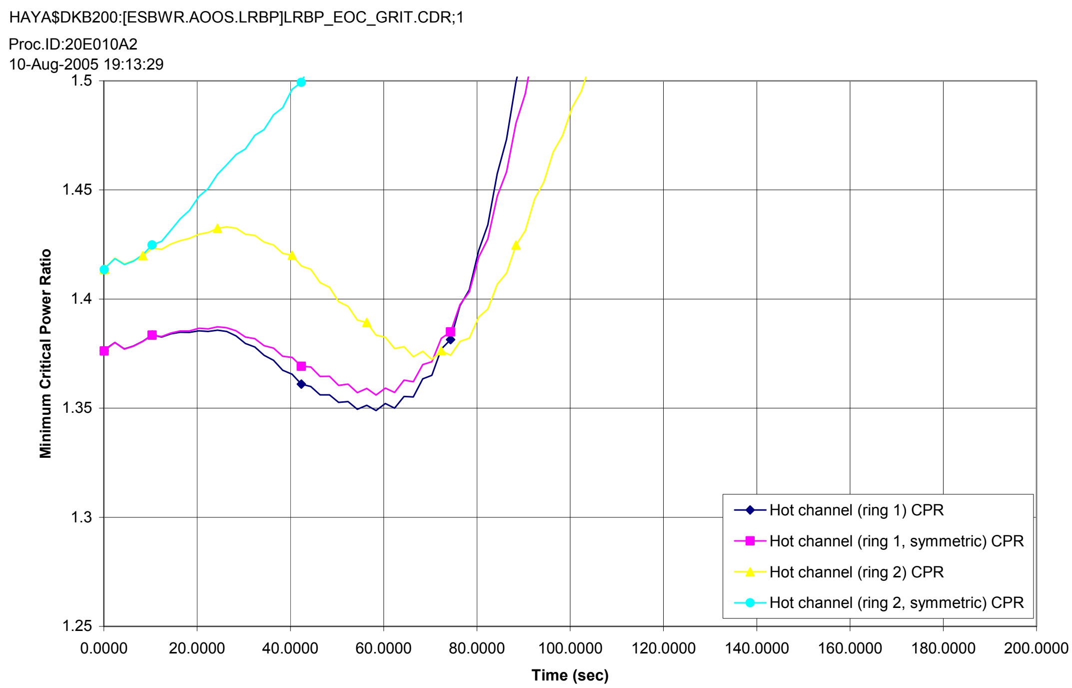


Figure 15.2-4g. Generator Load Rejection with Turbine Bypass

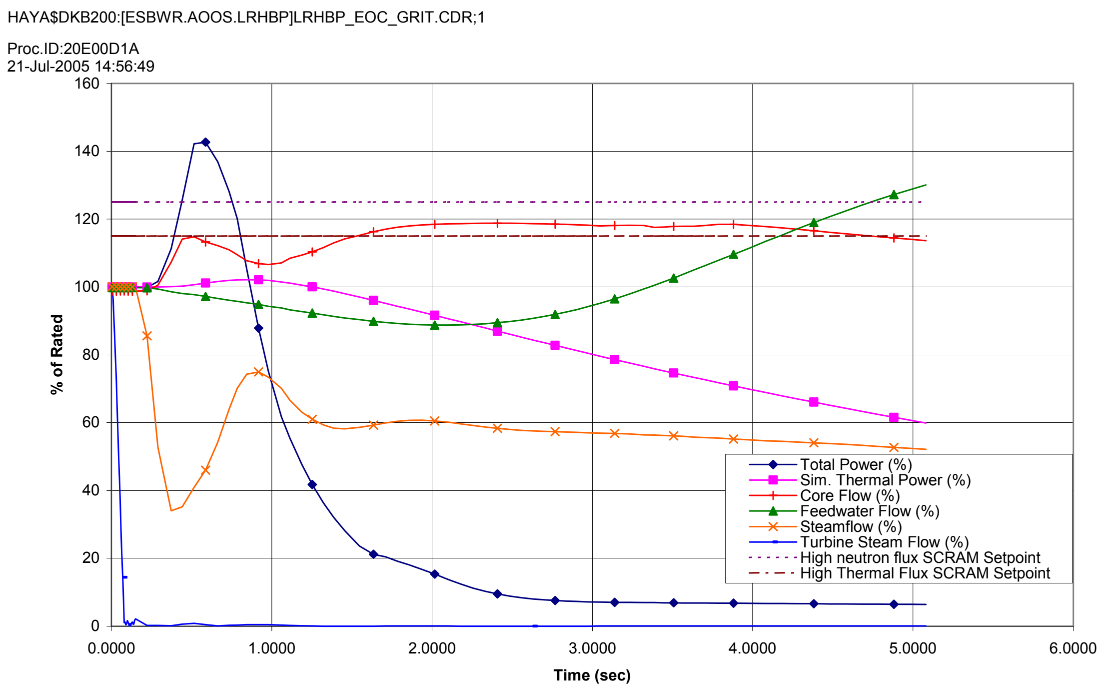


Figure 15.2-5a. Generator Load Rejection with a Single Failure in the Turbine Bypass System

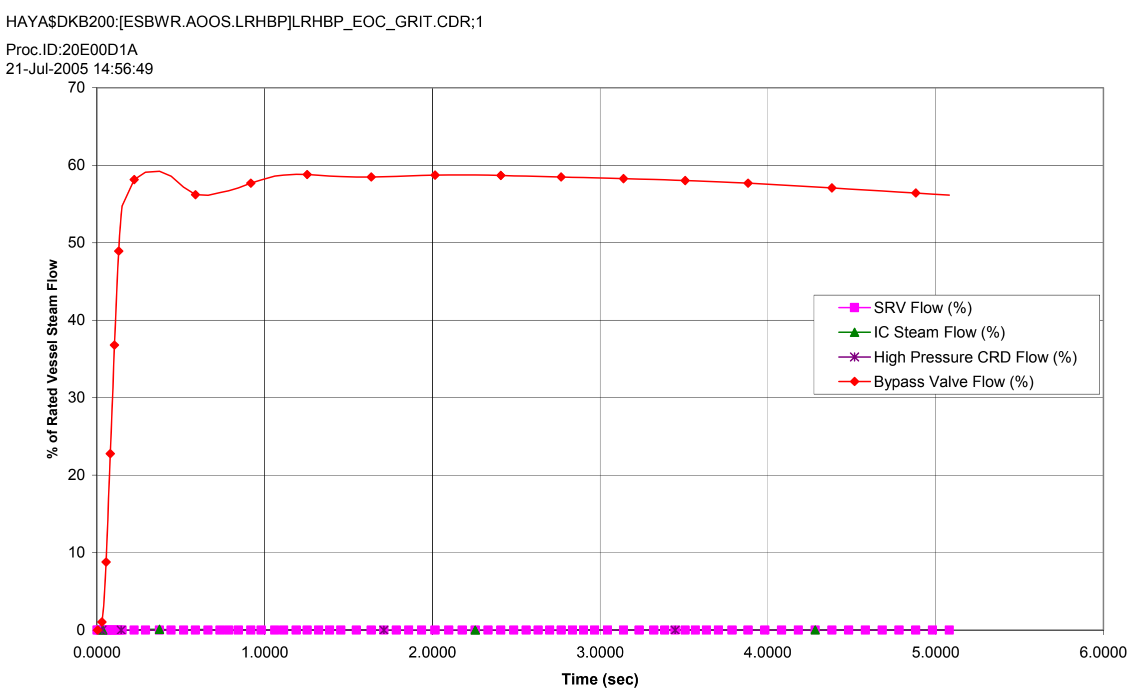


Figure 15.2-5b. Generator Load Rejection with a Single Failure in the Turbine Bypass System

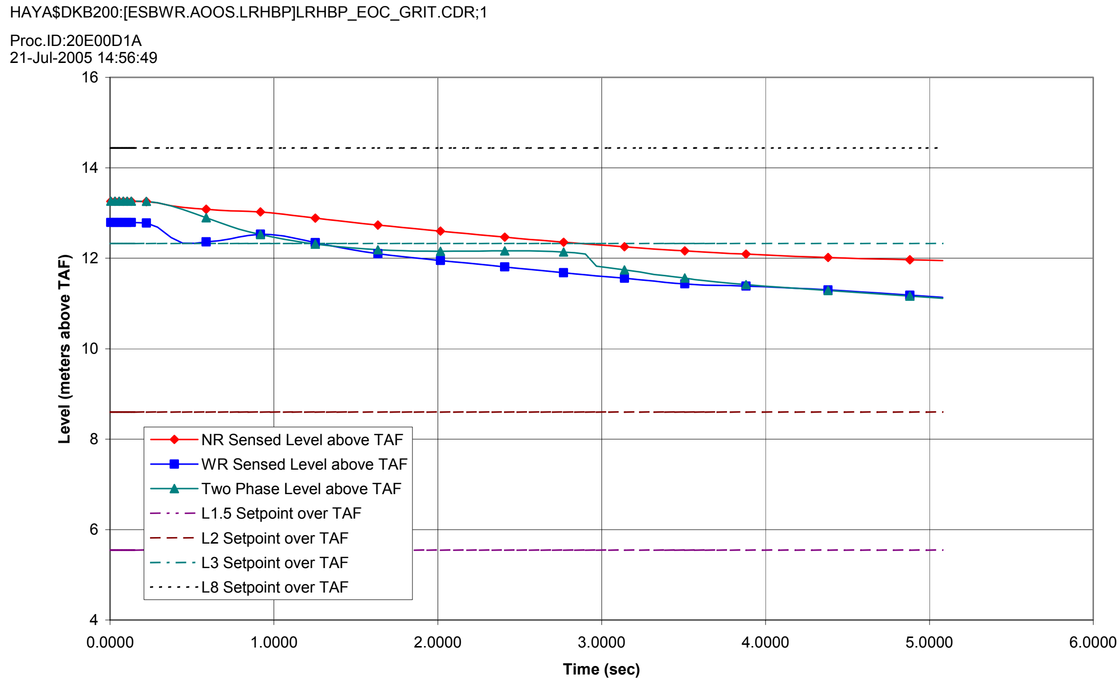


Figure 15.2-5c. Generator Load Rejection with a Single Failure in the Turbine Bypass System

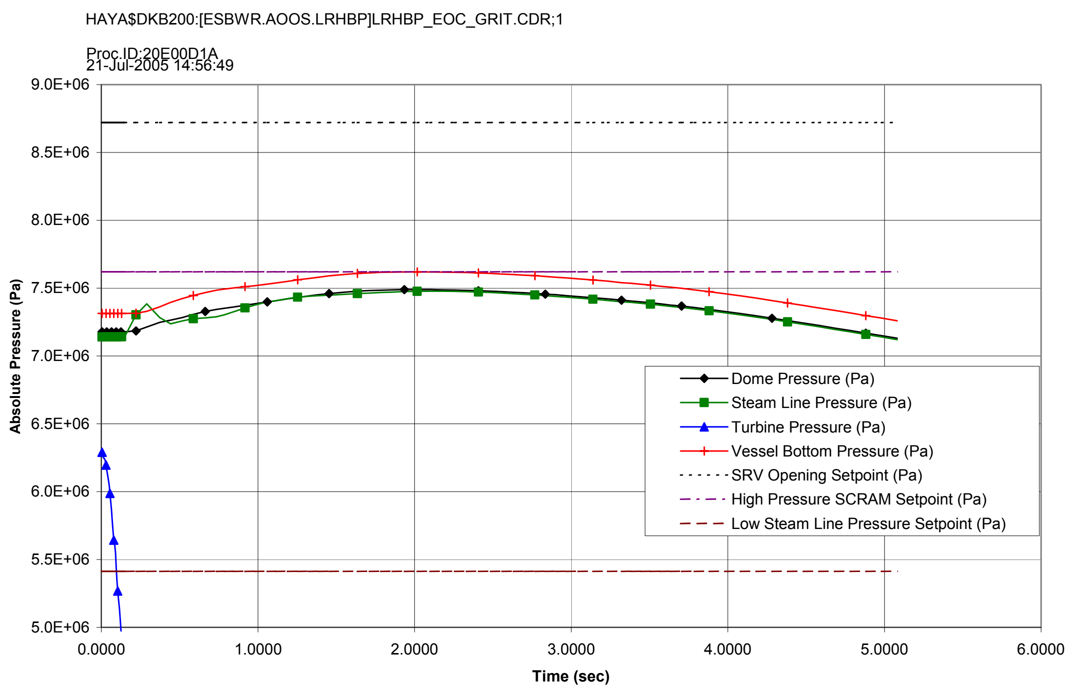


Figure 15.2-5d. Generator Load Rejection with a Single Failure in the Turbine Bypass System

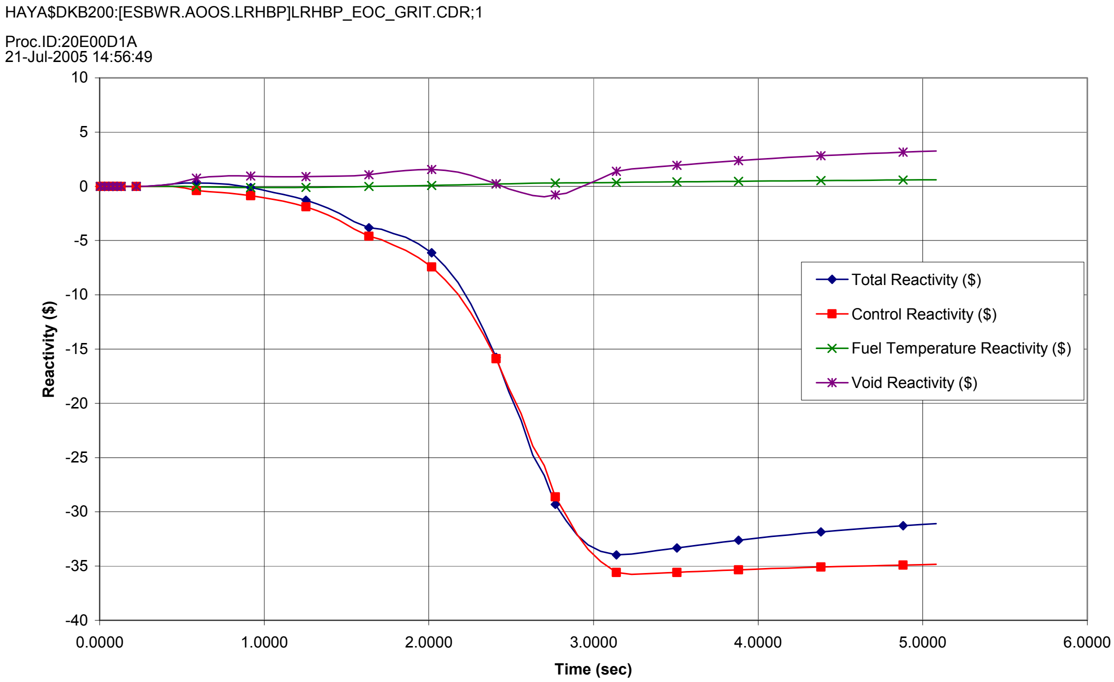


Figure 15.2-5e. Generator Load Rejection with a Single Failure in the Turbine Bypass System

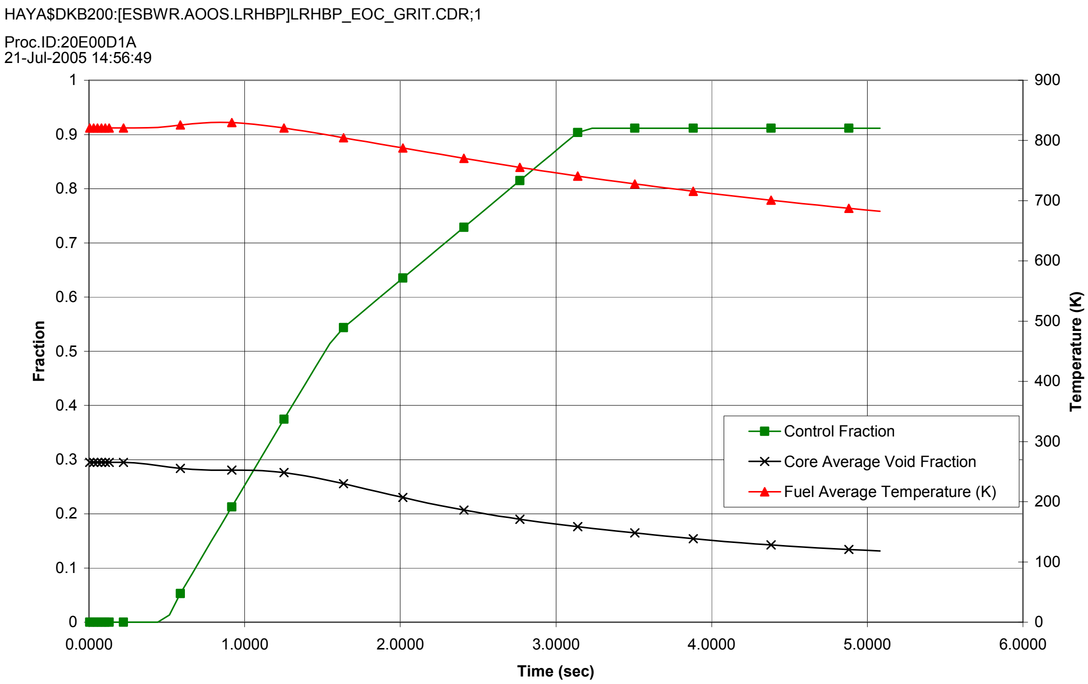


Figure 15.2-5f. Generator Load Rejection with a Single Failure in the Turbine Bypass System

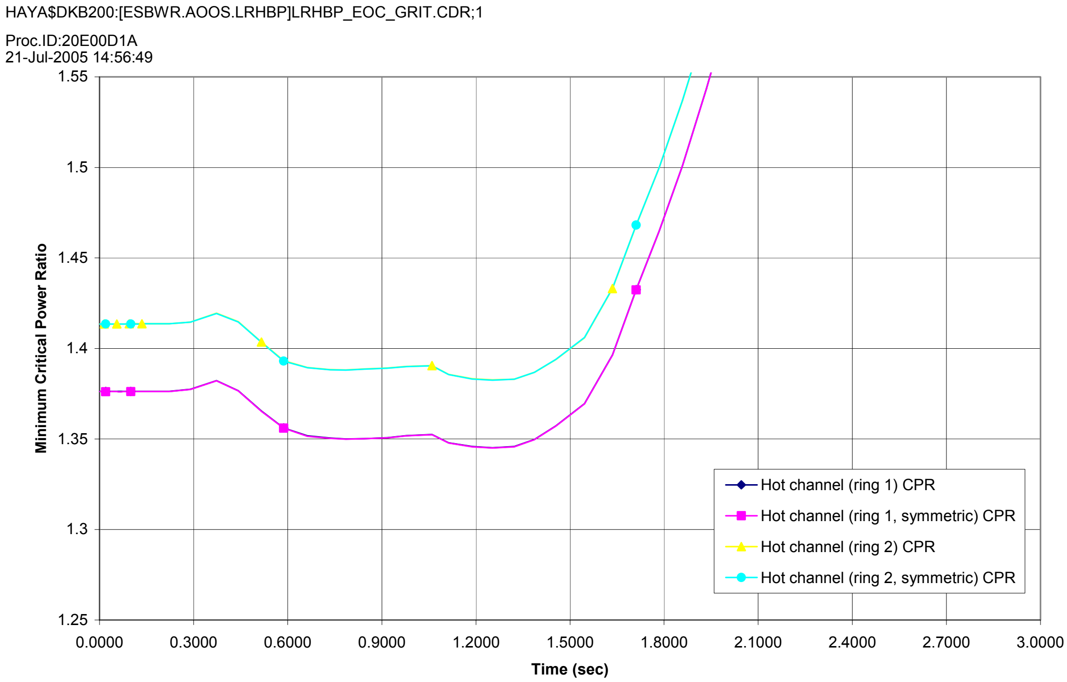


Figure 15.2-5g. Generator Load Rejection with a Single Failure in the Turbine Bypass System

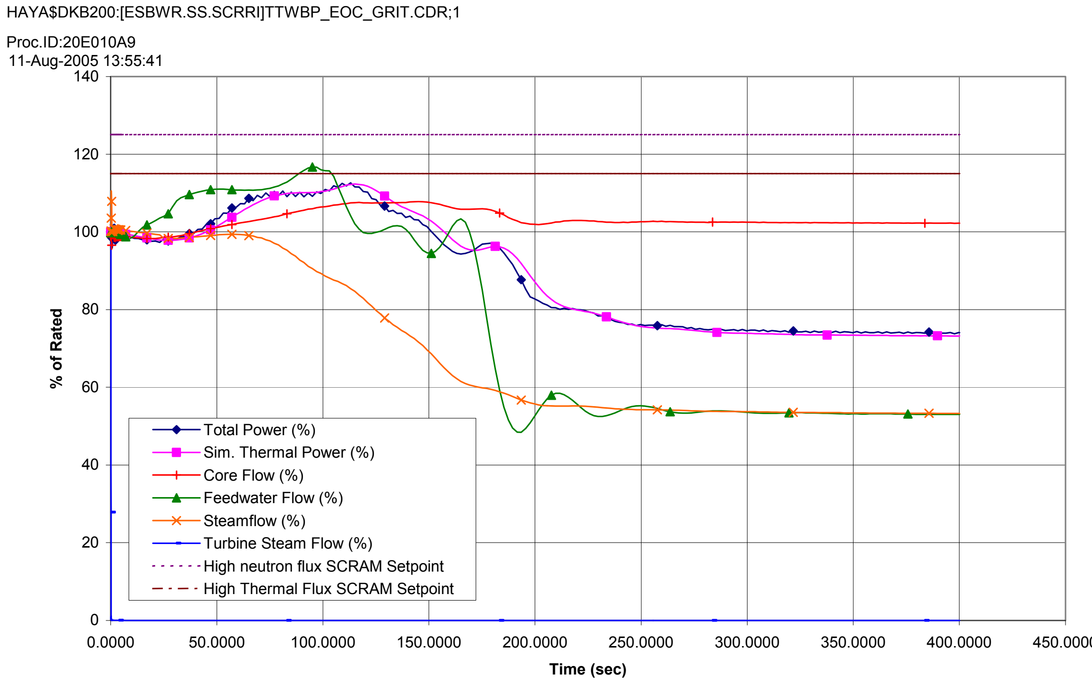


Figure 15.2-6a. Turbine Trip with Turbine Bypass

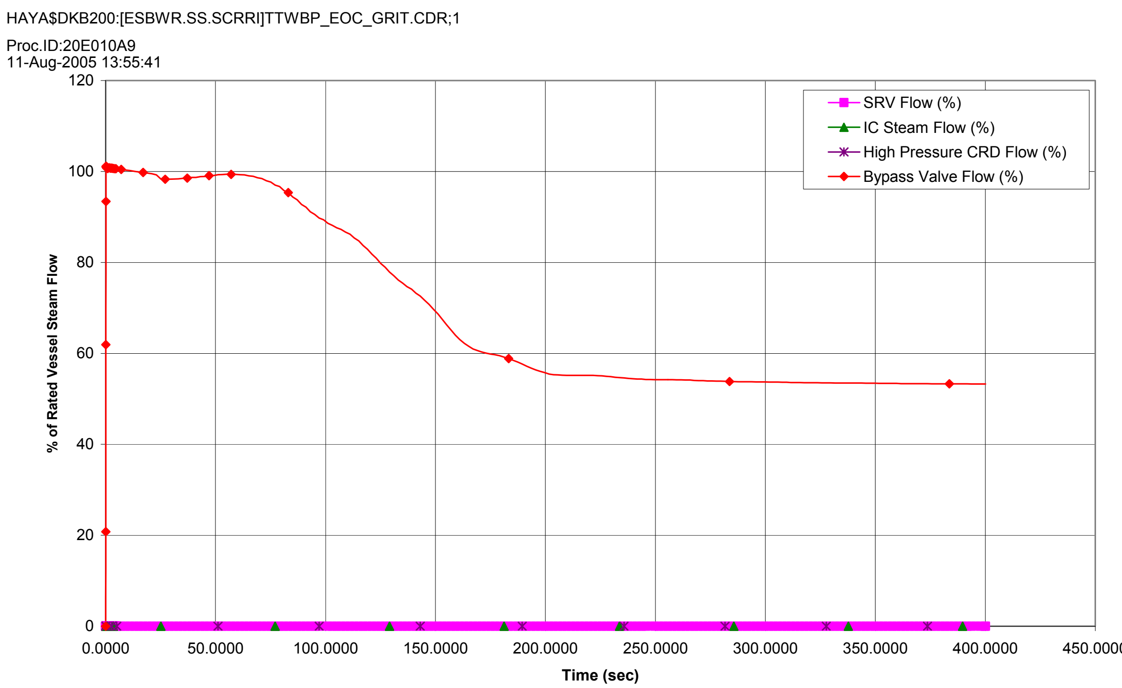


Figure 15.2-6b. Turbine Trip with Turbine Bypass

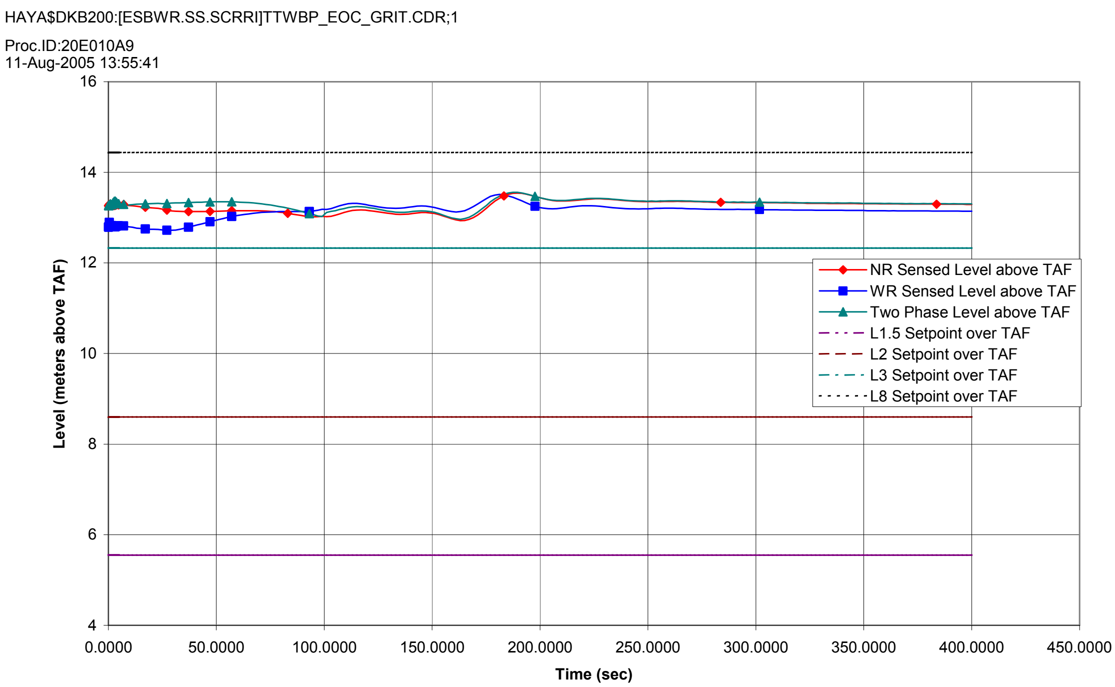


Figure 15.2-6c. Turbine Trip with Turbine Bypass

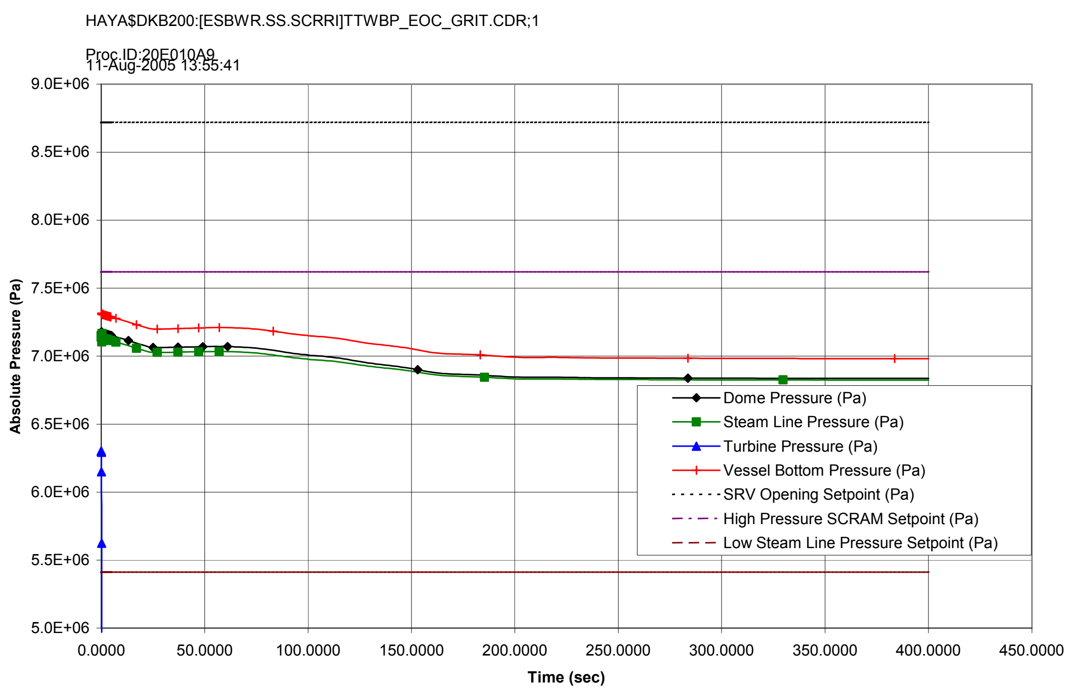


Figure 15.2-6d. Turbine Trip with Turbine Bypass

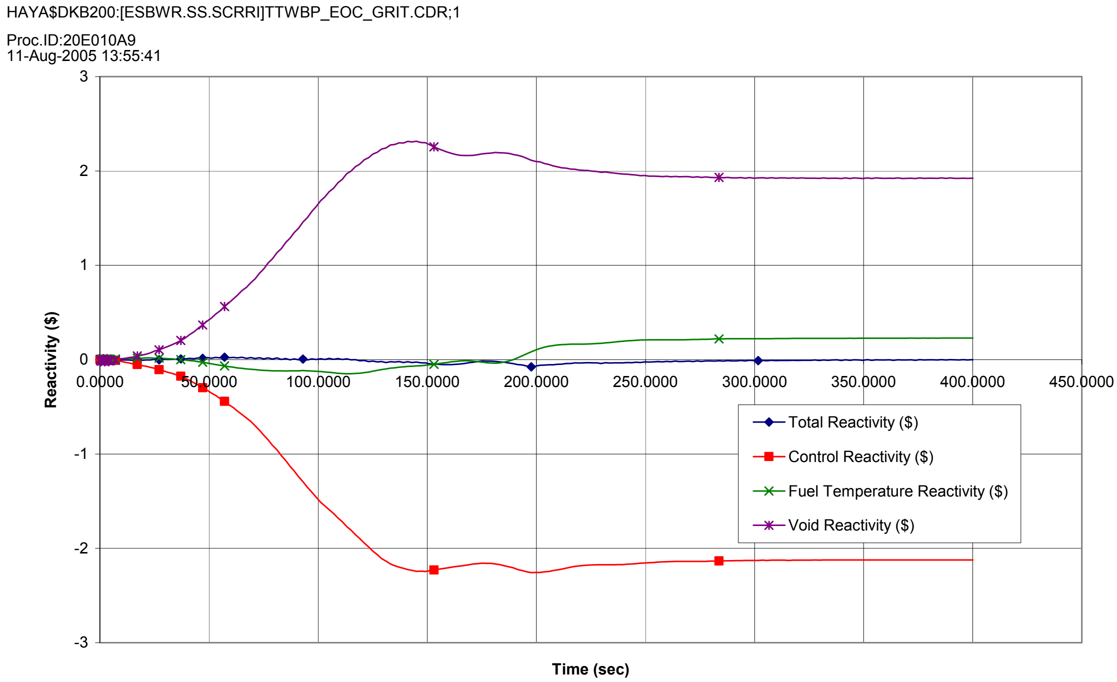


Figure 15.2-6e. Turbine Trip with Turbine Bypass

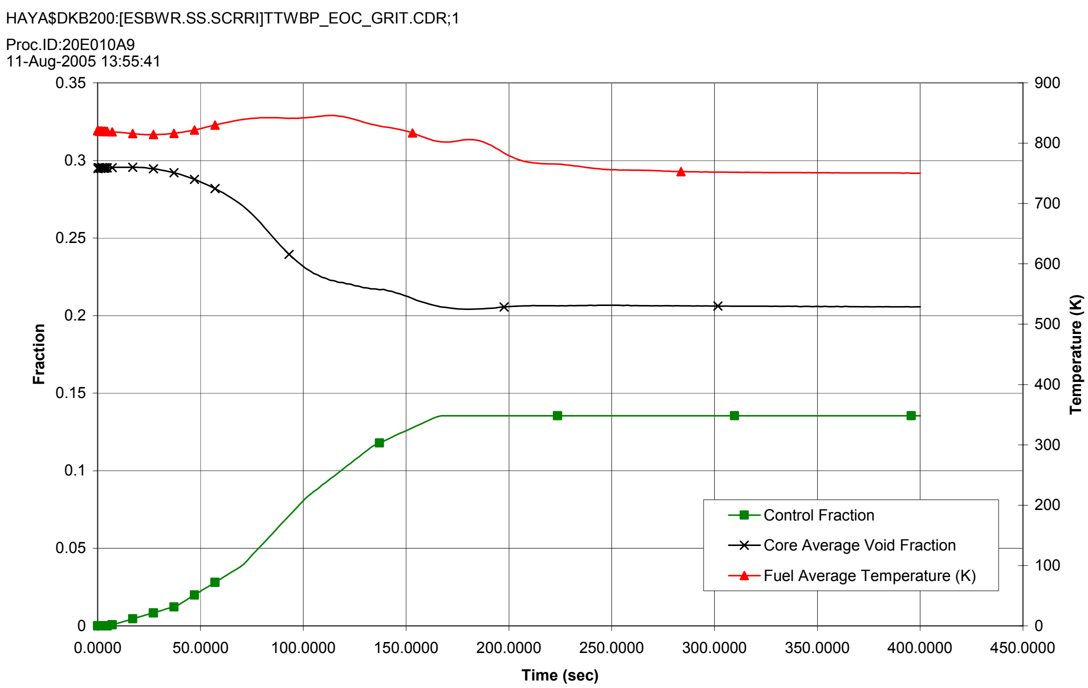


Figure 15.2-6f. Turbine Trip with Turbine Bypass

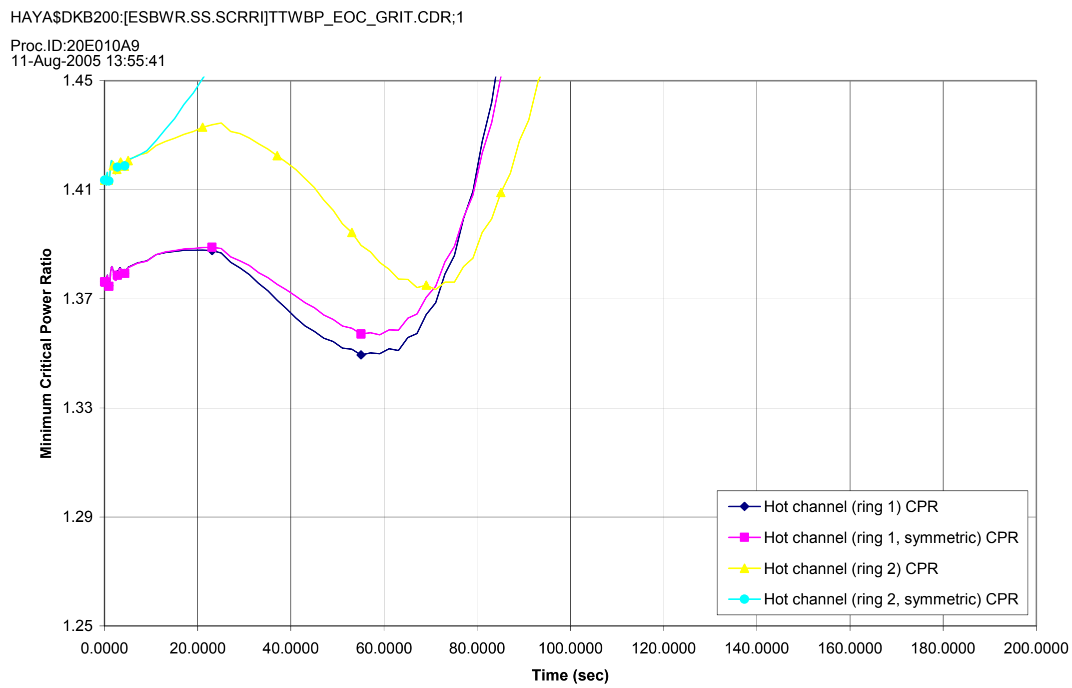


Figure 15.2-6g. Turbine Trip with Turbine Bypass

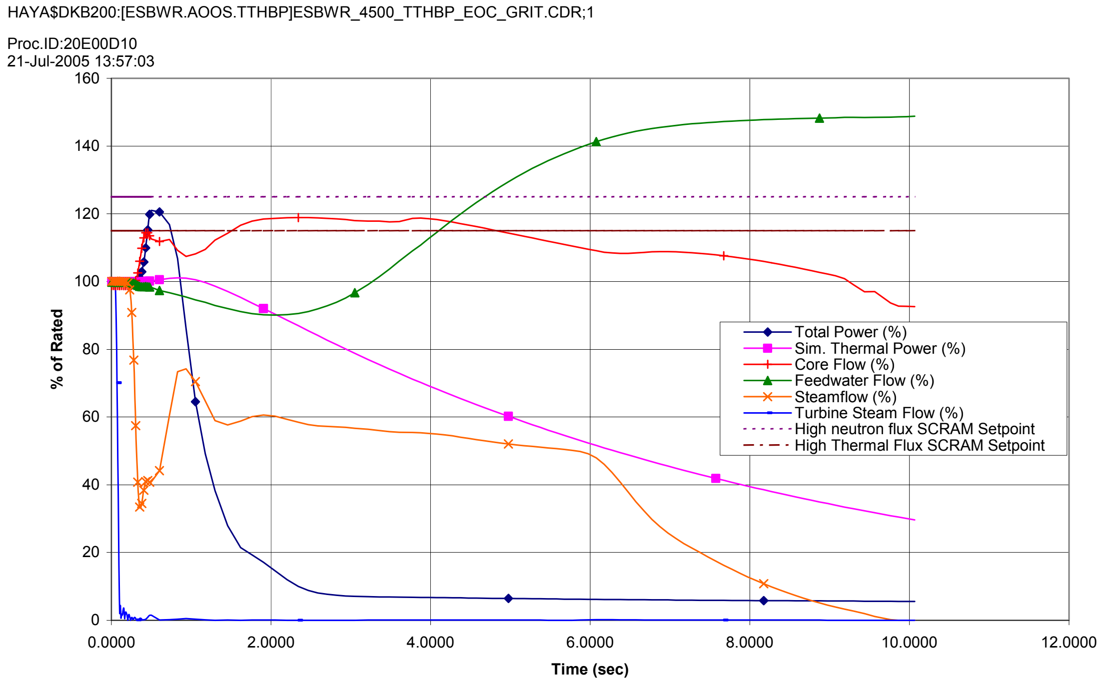


Figure 15.2-7a. Turbine Trip with a Single Failure in the Turbine Bypass System

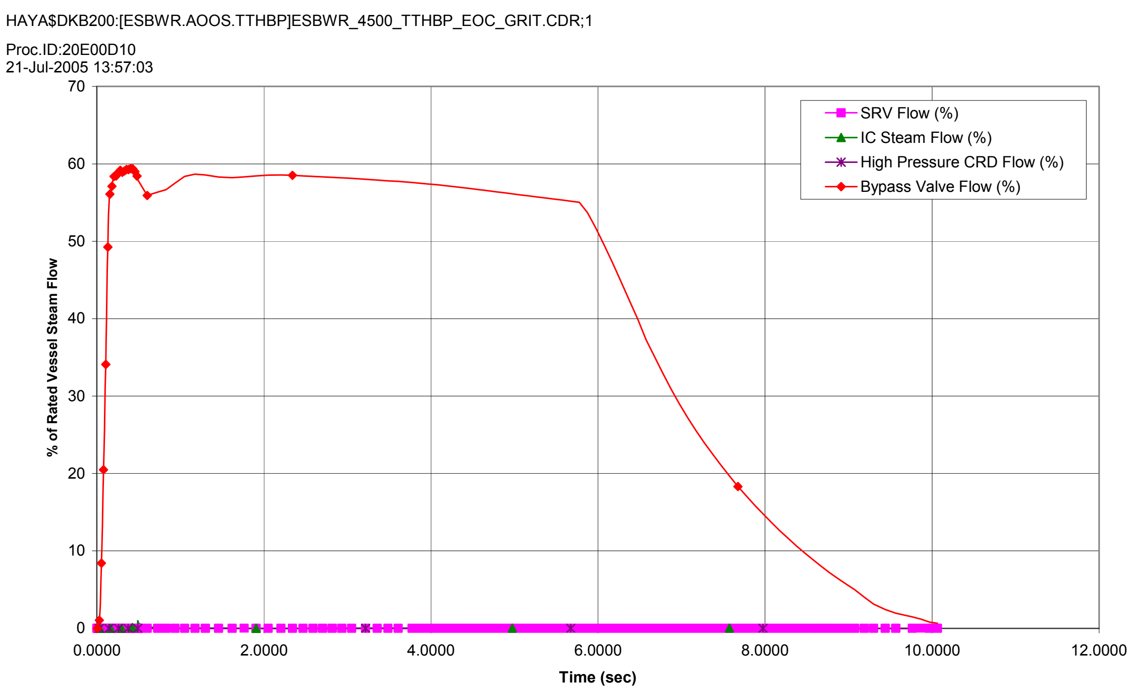


Figure 15.2-7b. Turbine Trip with a Single Failure in the Turbine Bypass System

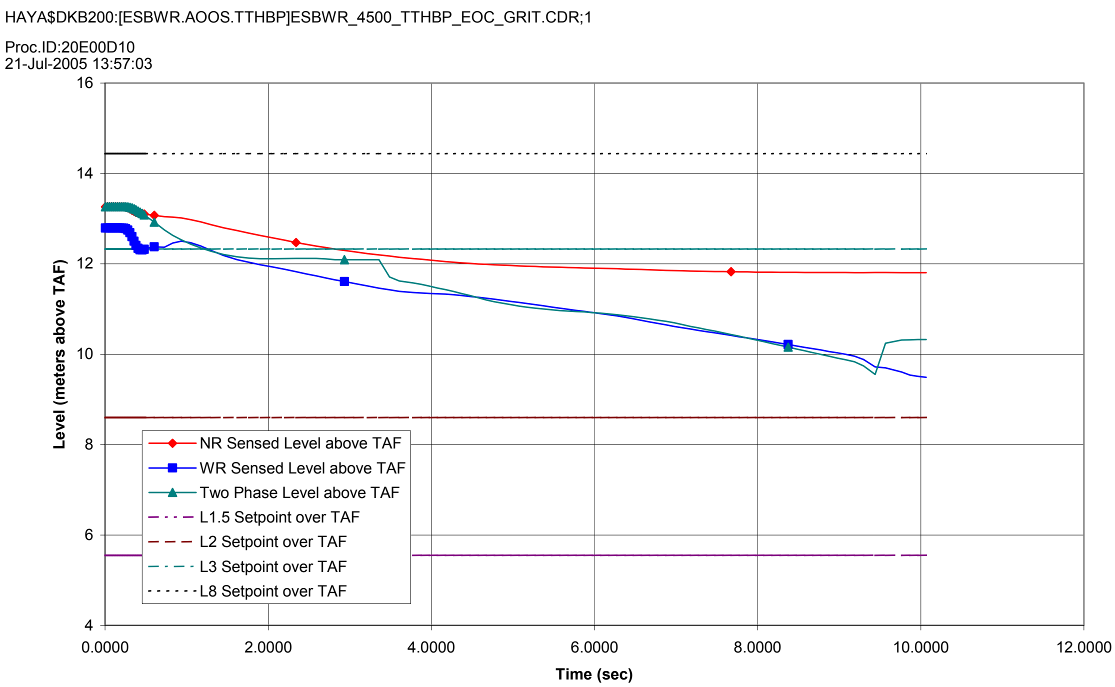


Figure 15.2-7c. Turbine Trip with a Single Failure in the Turbine Bypass System

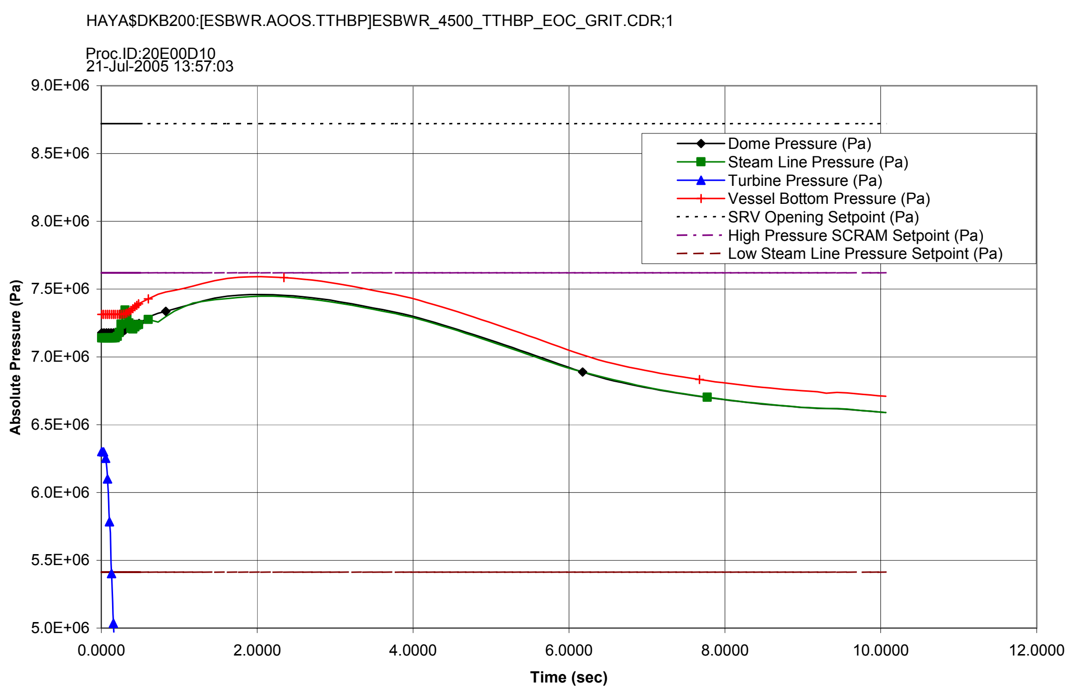


Figure 15.2-7d. Turbine Trip with a Single Failure in the Turbine Bypass System

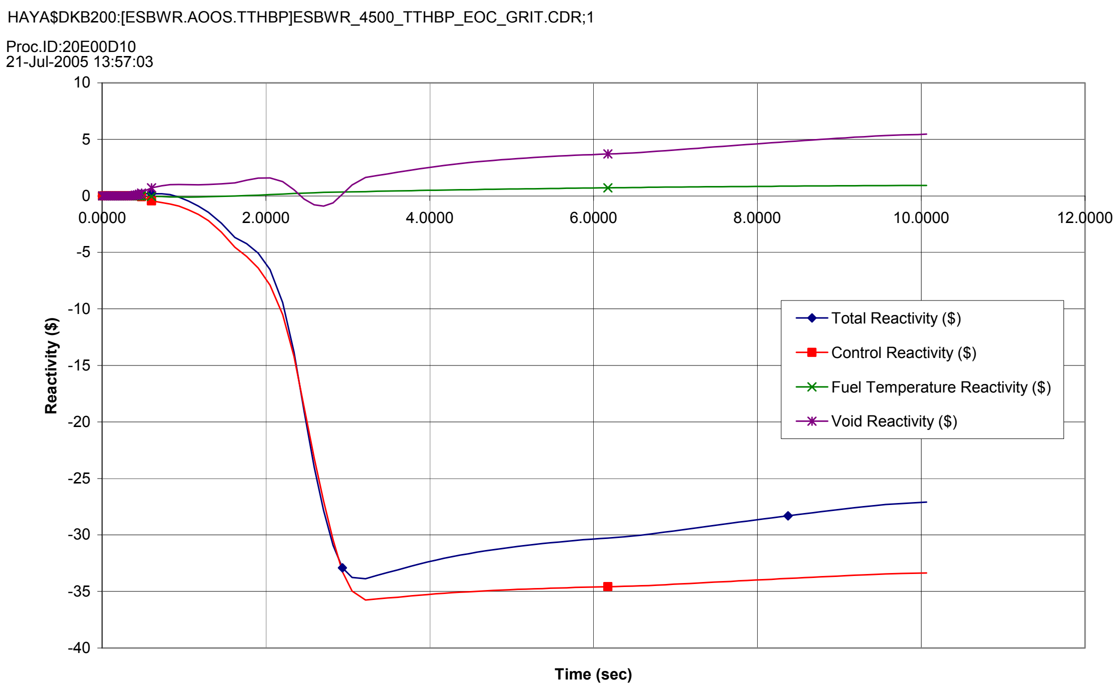


Figure 15.2-7e. Turbine Trip with a Single Failure in the Turbine Bypass System

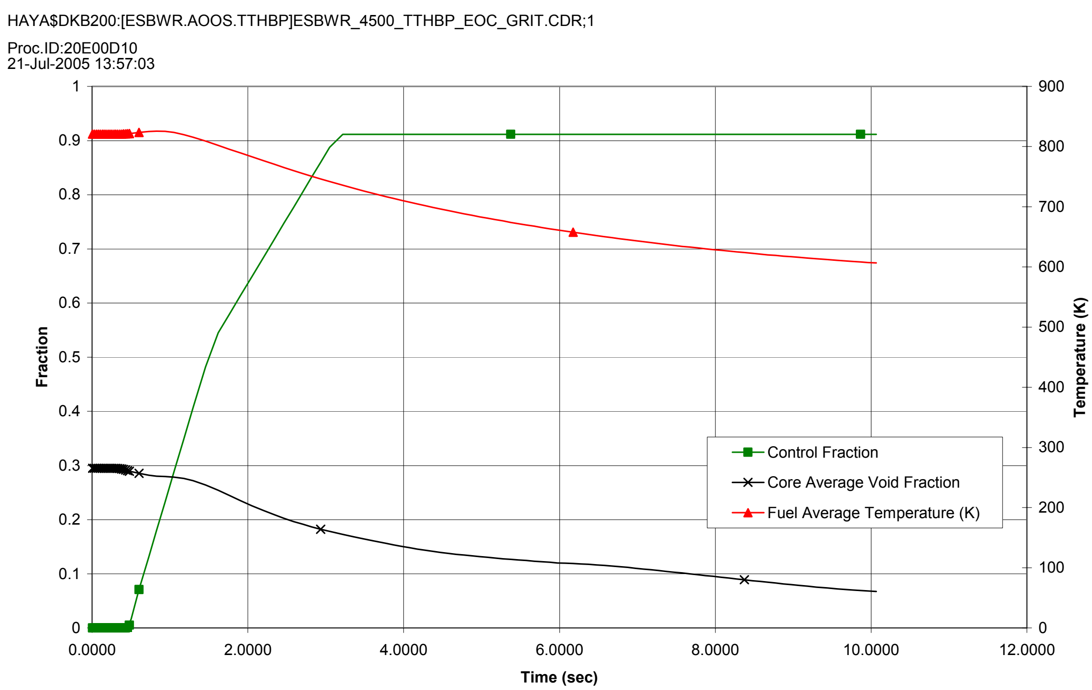


Figure 15.2-7f. Turbine Trip with a Single Failure in the Turbine Bypass System

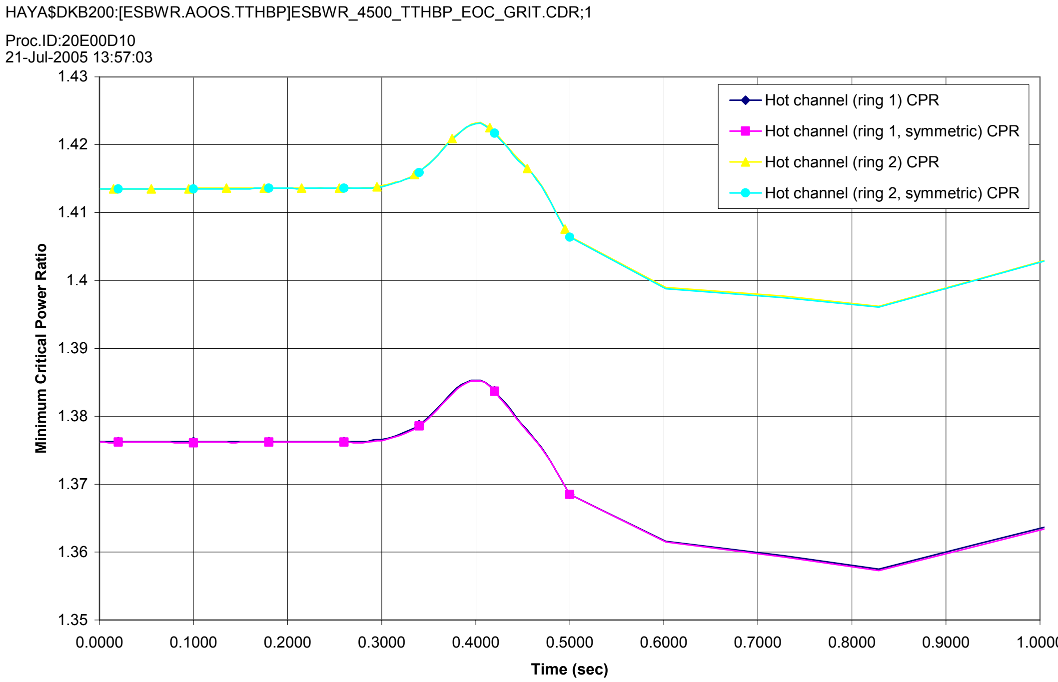


Figure 15.2-7g. Turbine Trip with a Single Failure in the Turbine Bypass System

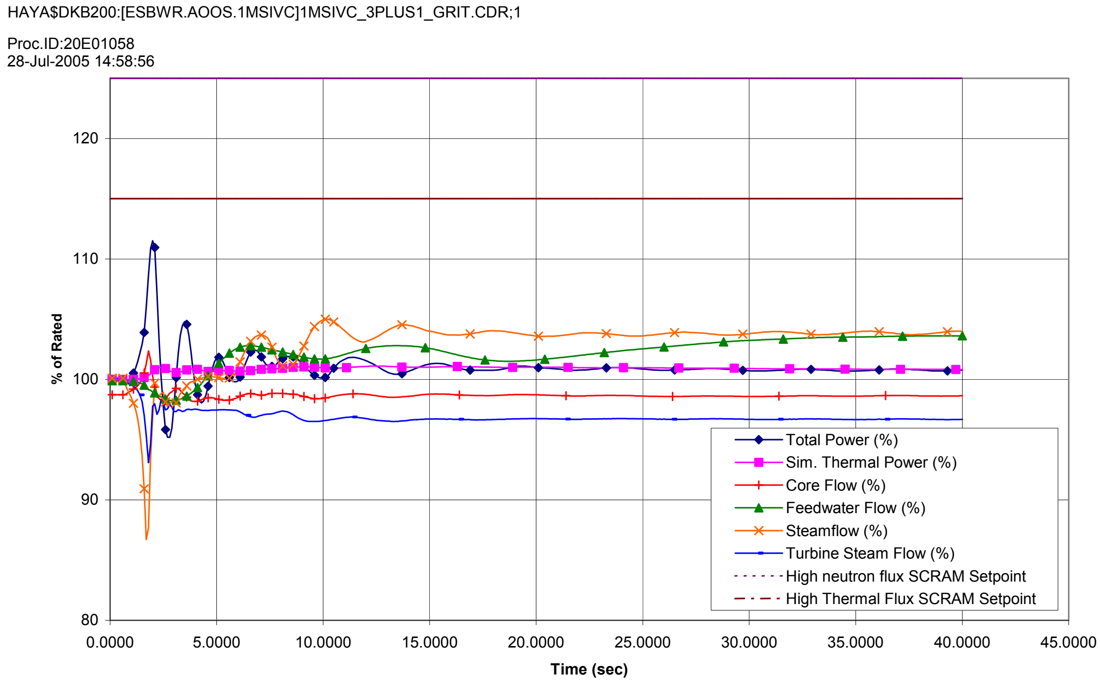


Figure 15.2-8a. One MSIV Closure

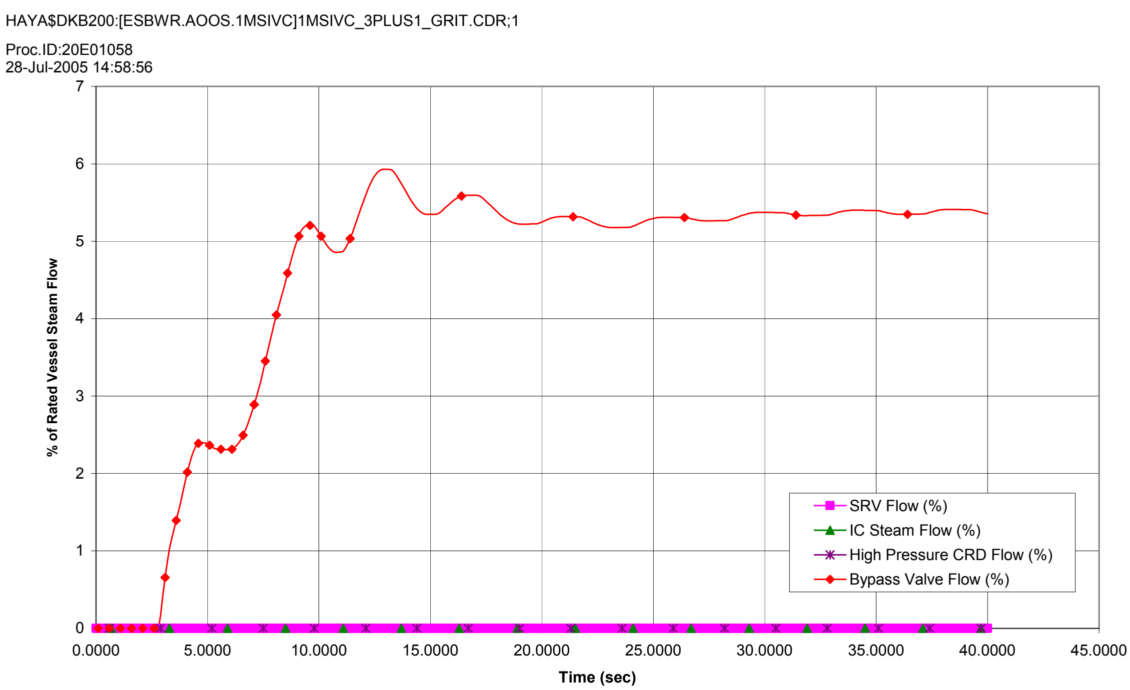


Figure 15.2-8b. One MSIV Closure

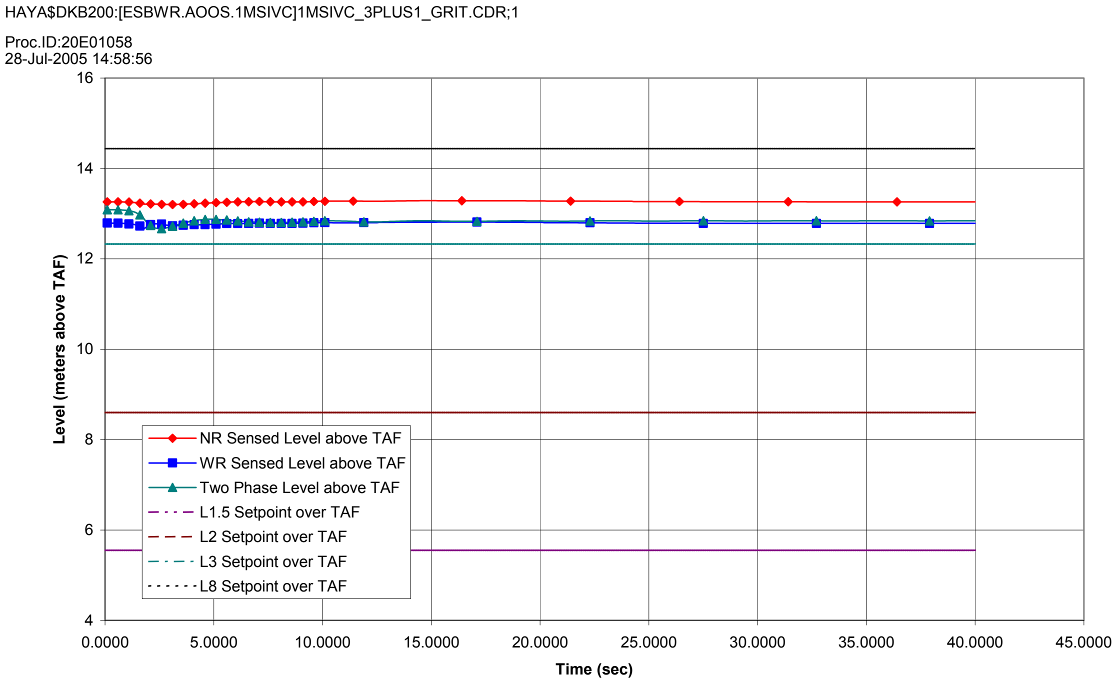


Figure 15.2-8c. One MSIV Closure

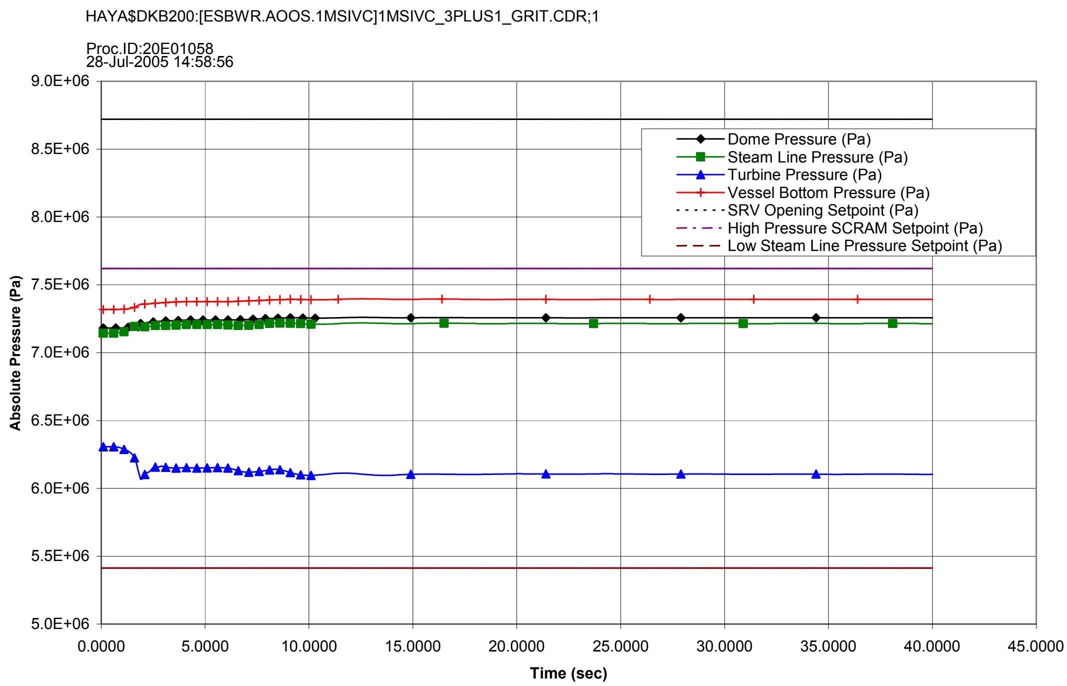


Figure 15.2-8d. One MSIV Closure

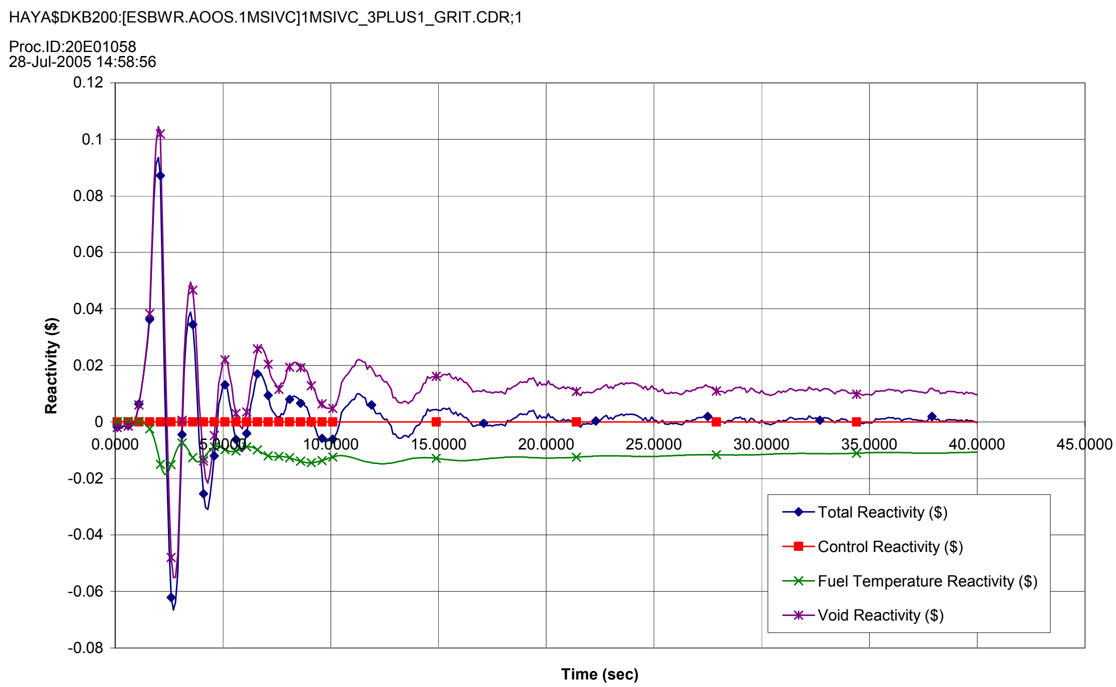


Figure 15.2-8e. One MSIV Closure

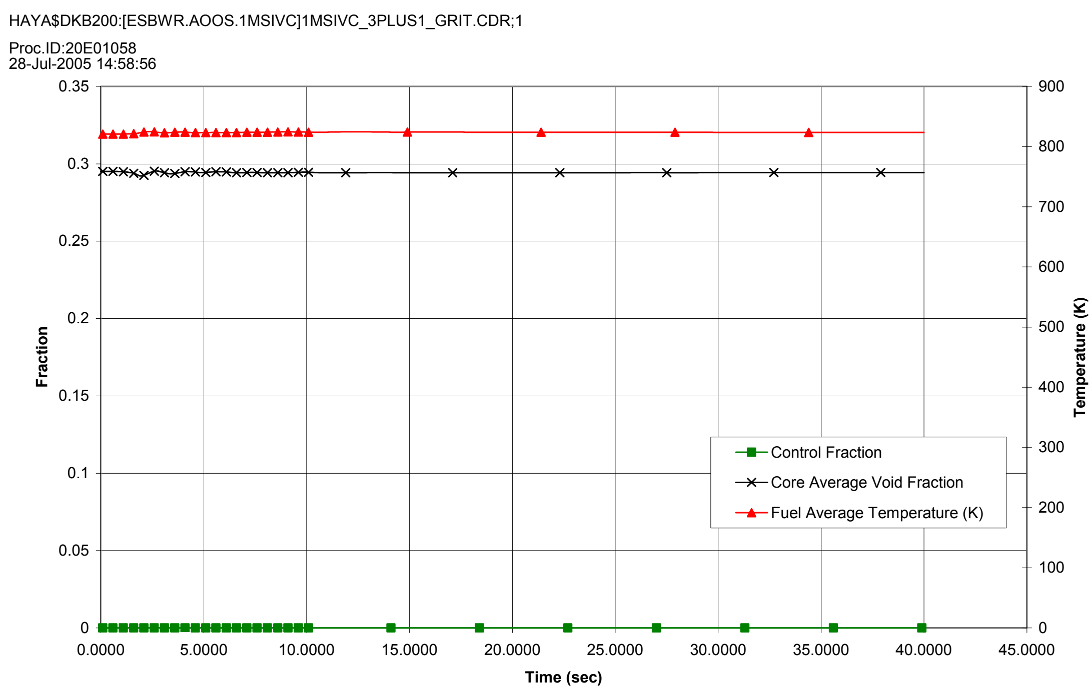


Figure 15.2-8f. One MSIV Closure

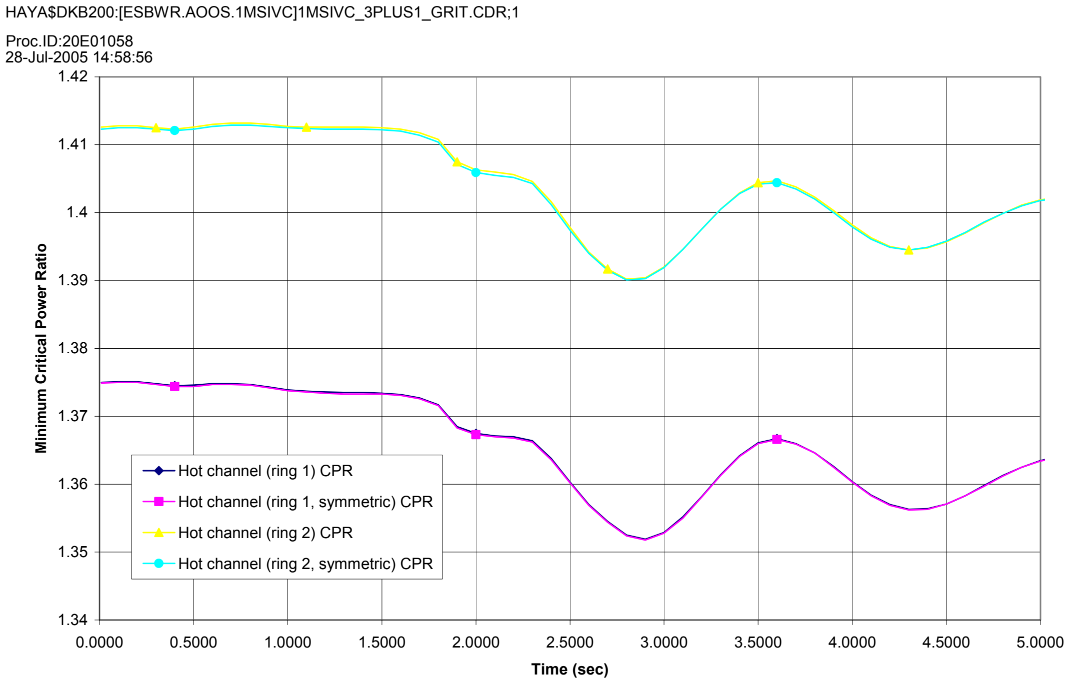


Figure 15.2-8g. One MSIV Closure

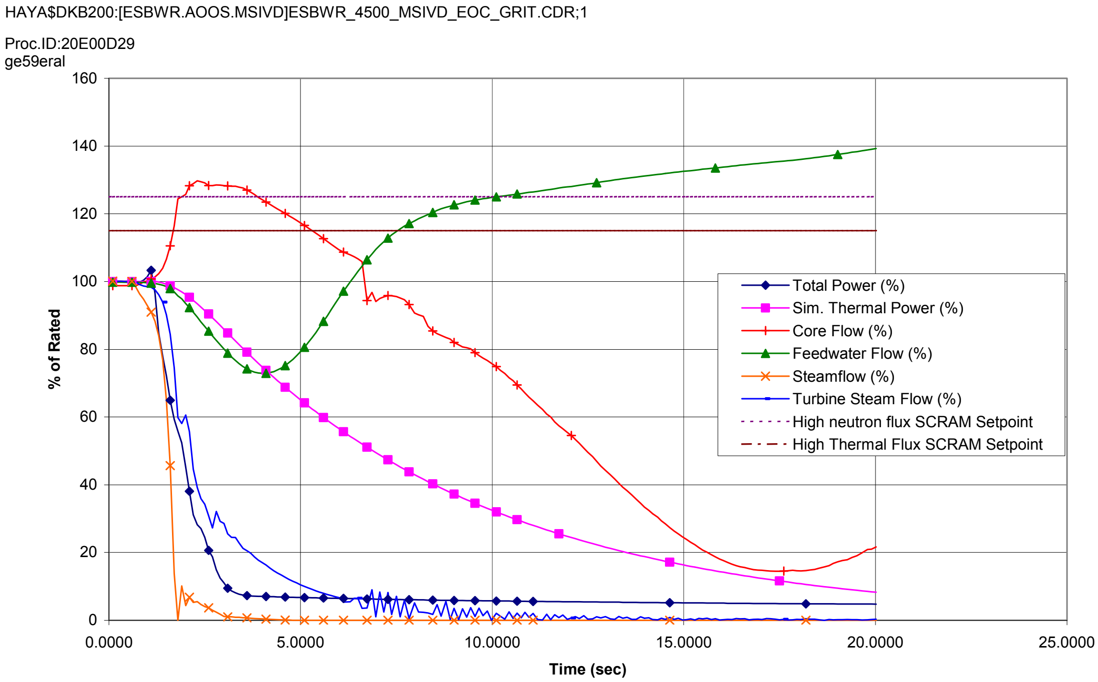


Figure 15.2-9a. MSIV Closure

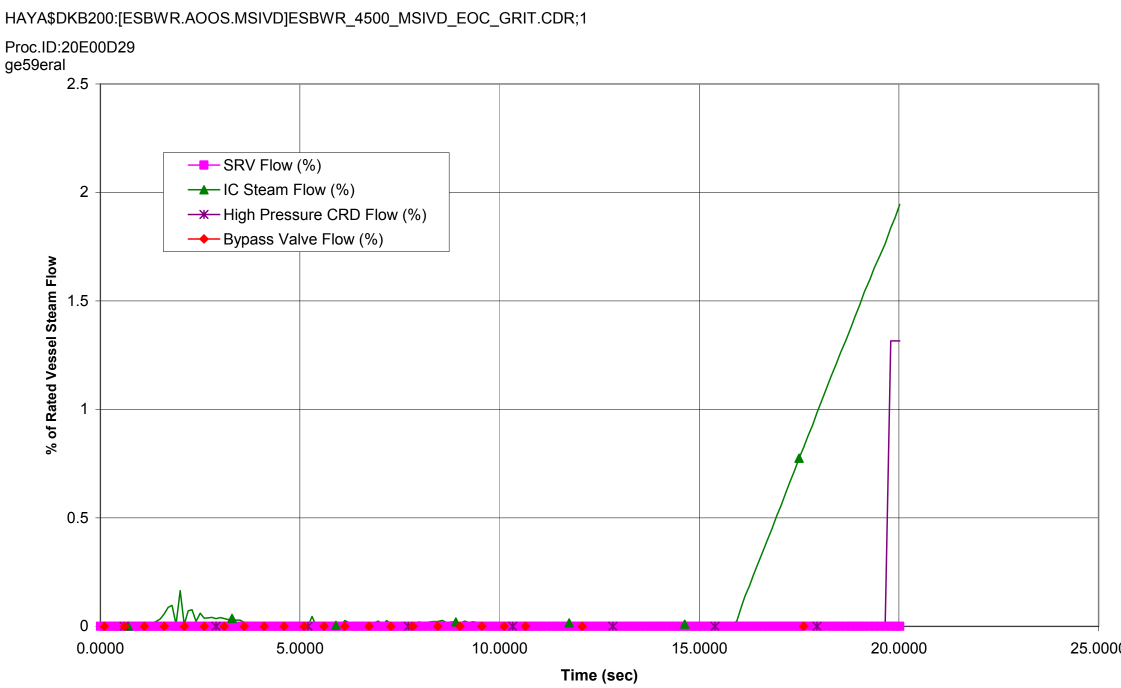


Figure 15.2-9b. MSIV Closure

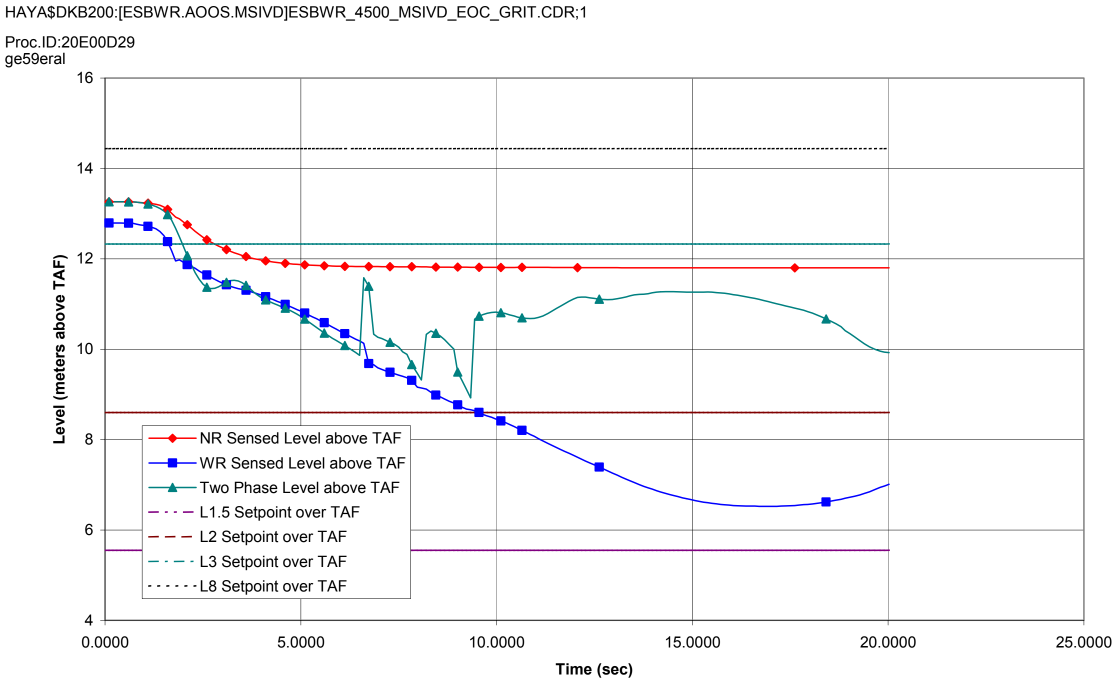


Figure 15.2-9c. MSIV Closure

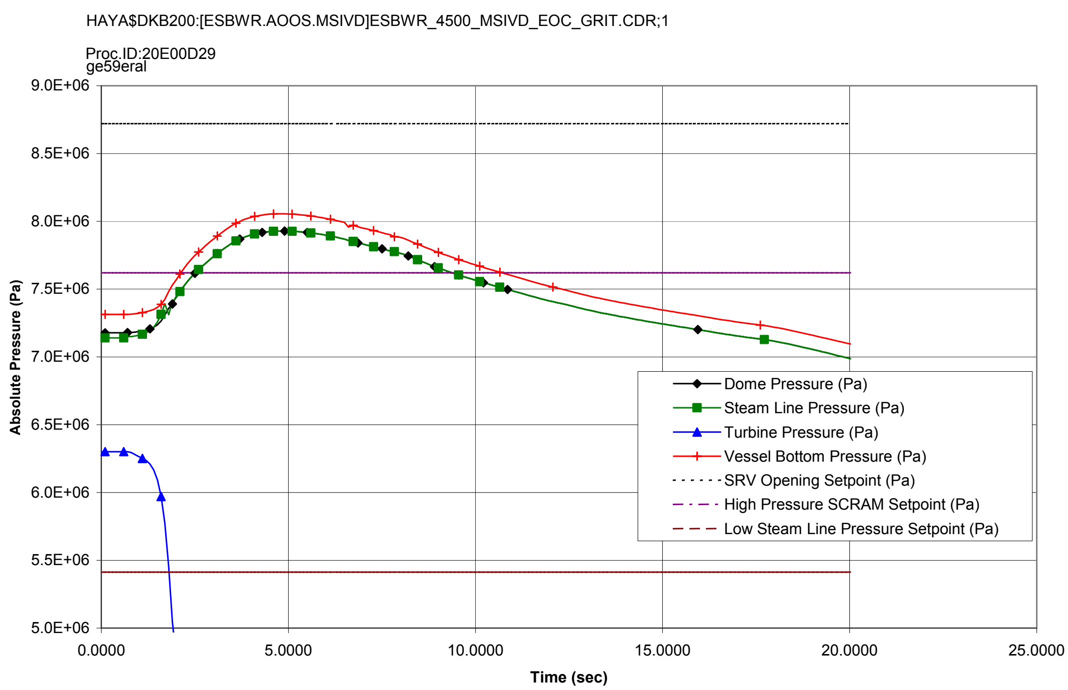


Figure 15.2-9d. MSIV Closure

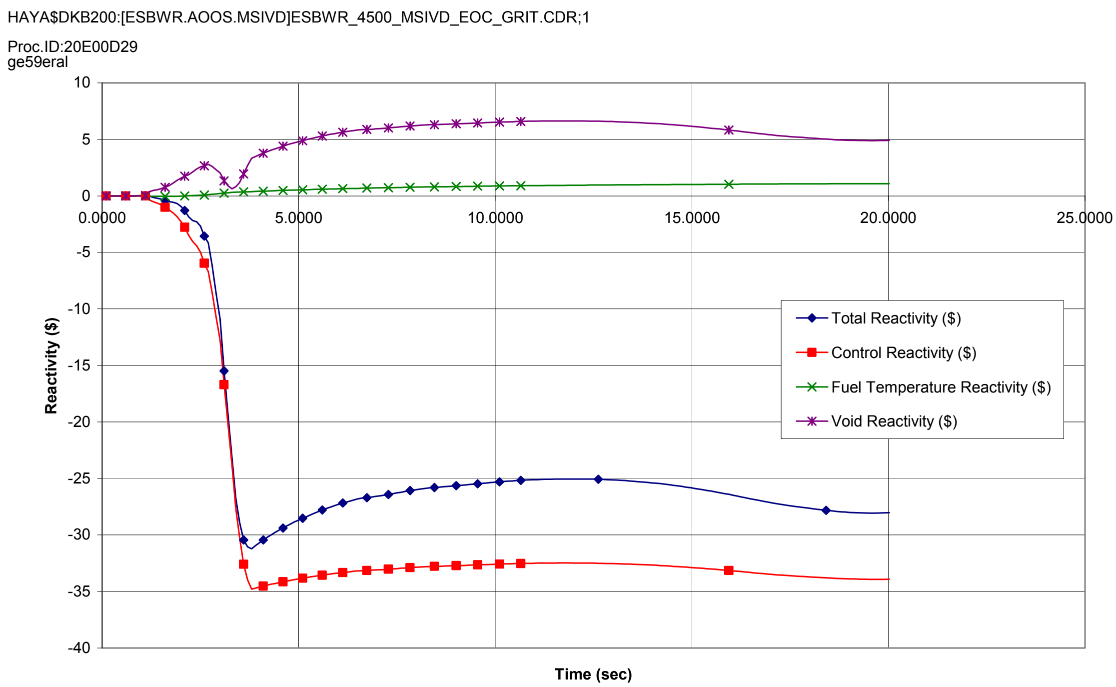


Figure 15.2-9e. MSIV Closure

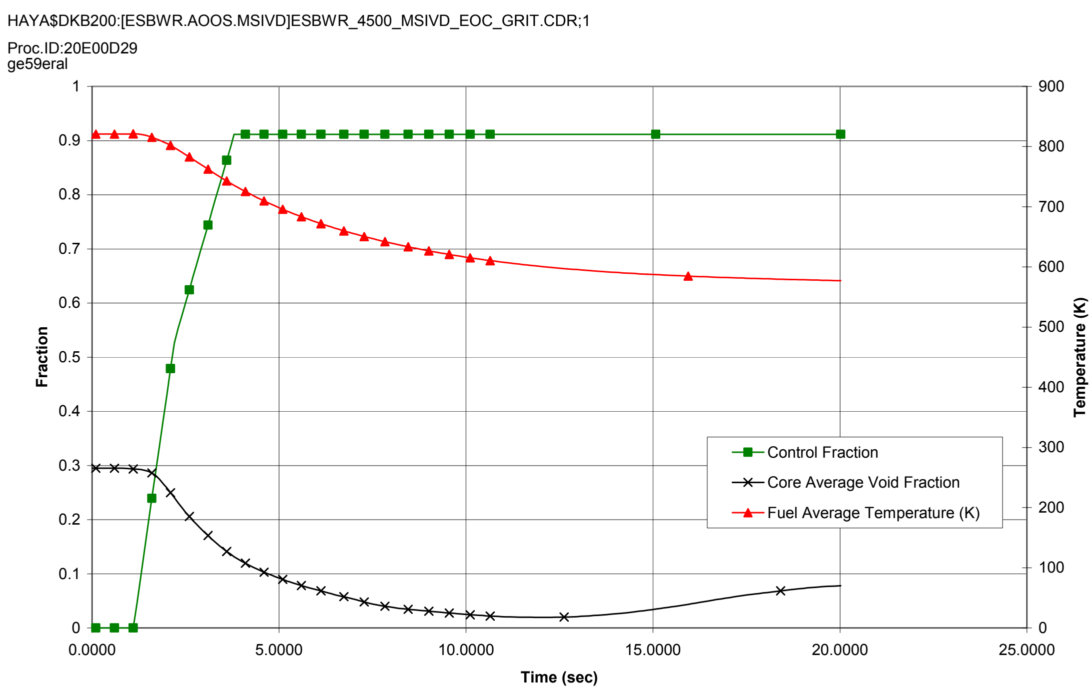


Figure 15.2-9f. MSIV Closure

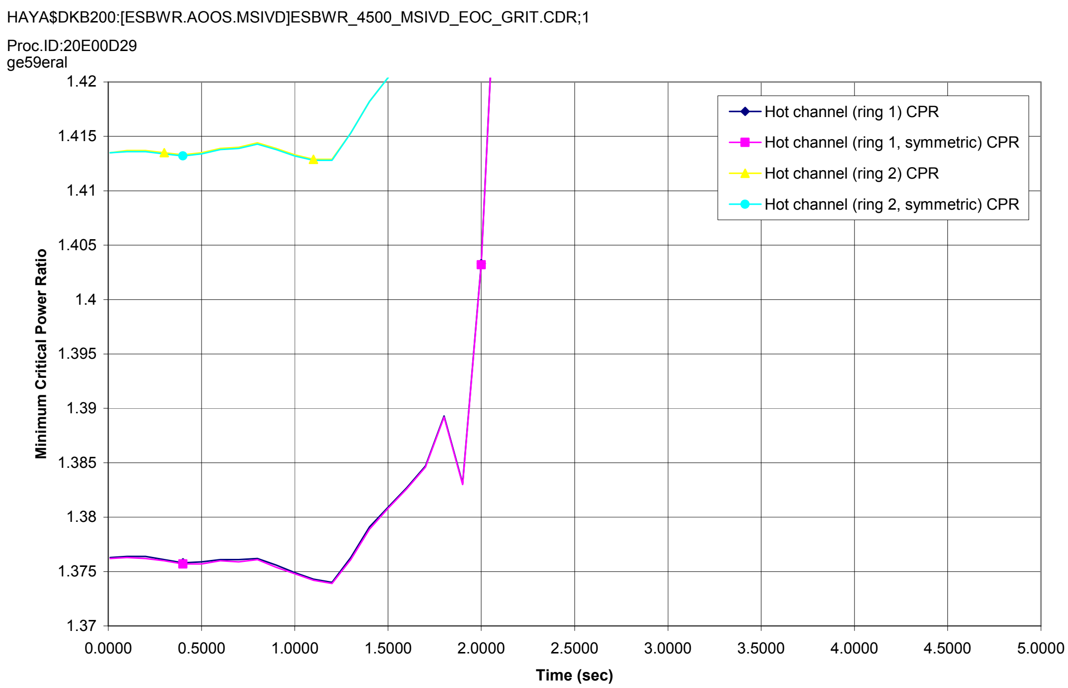


Figure 15.2-9g. MSIV Closure

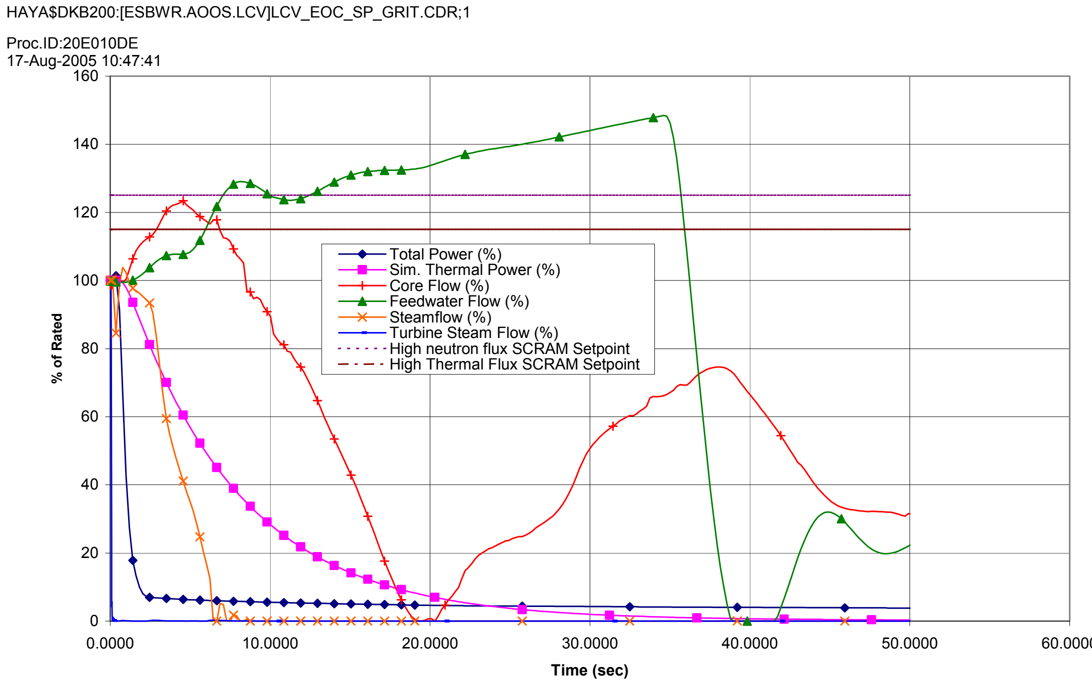


Figure 15.2-10a. Loss of Condenser Vacuum

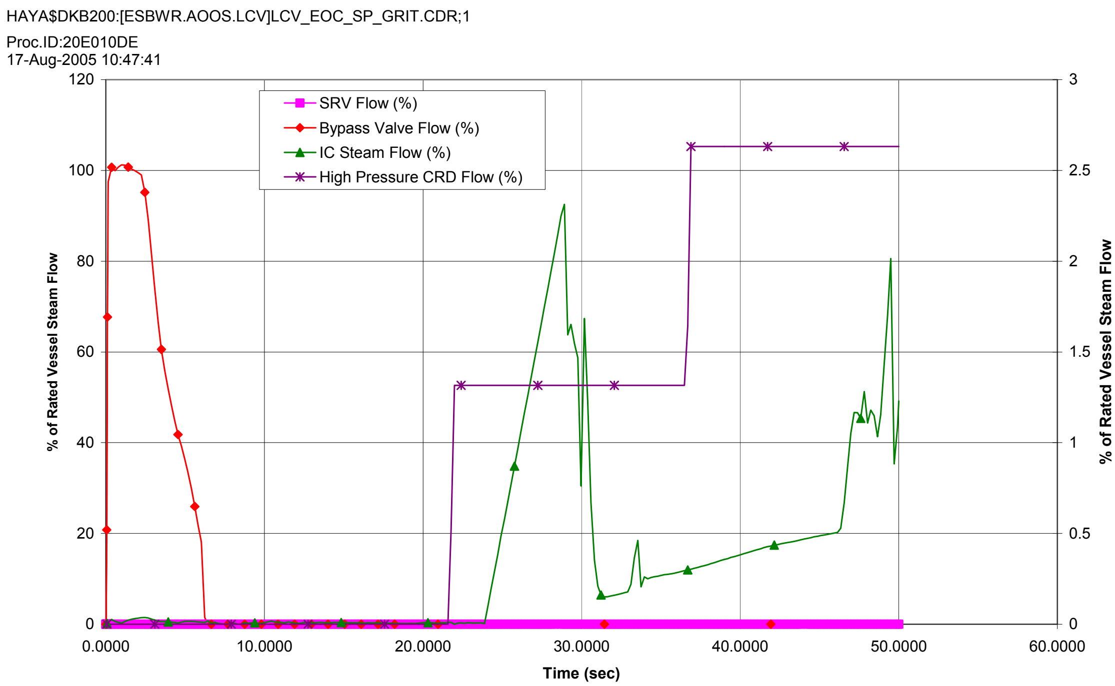


Figure 15.2-10b. Loss of Condenser Vacuum

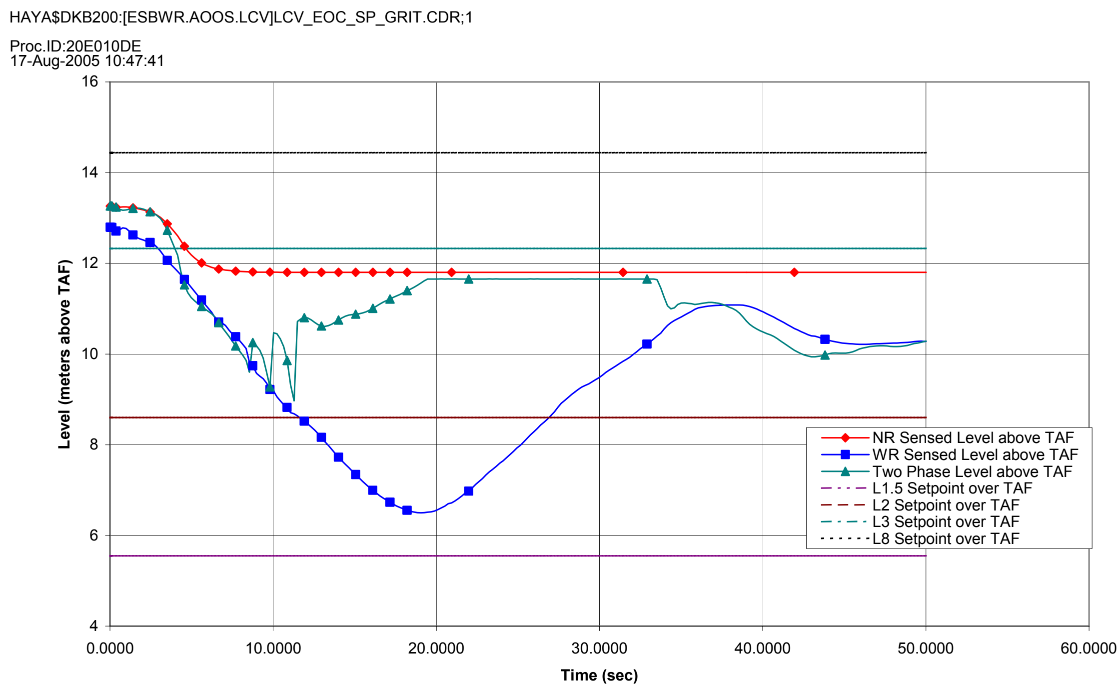


Figure 15.2-10c. Loss of Condenser Vacuum

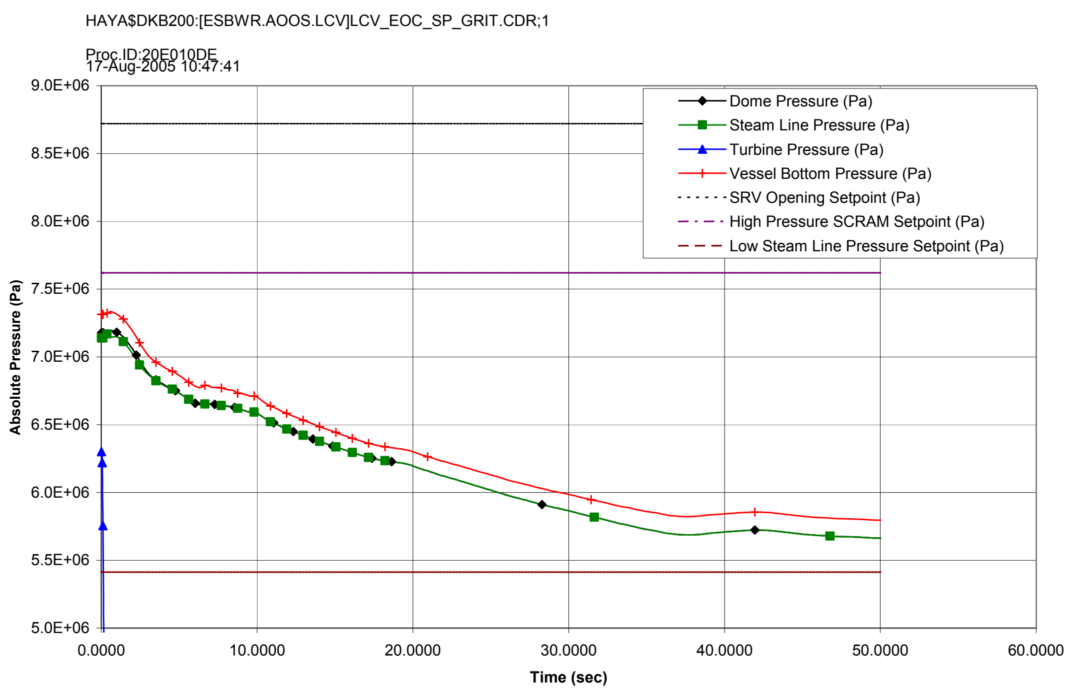


Figure 15.2-10d. Loss of Condenser Vacuum

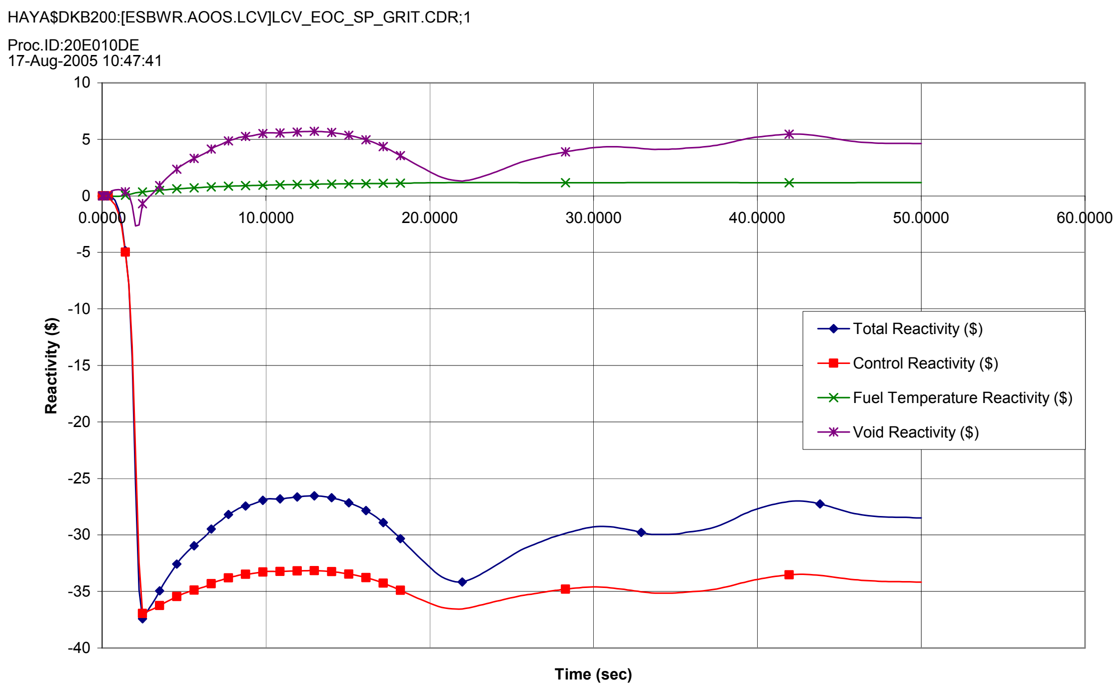


Figure 15.2-10e. Loss of Condenser Vacuum

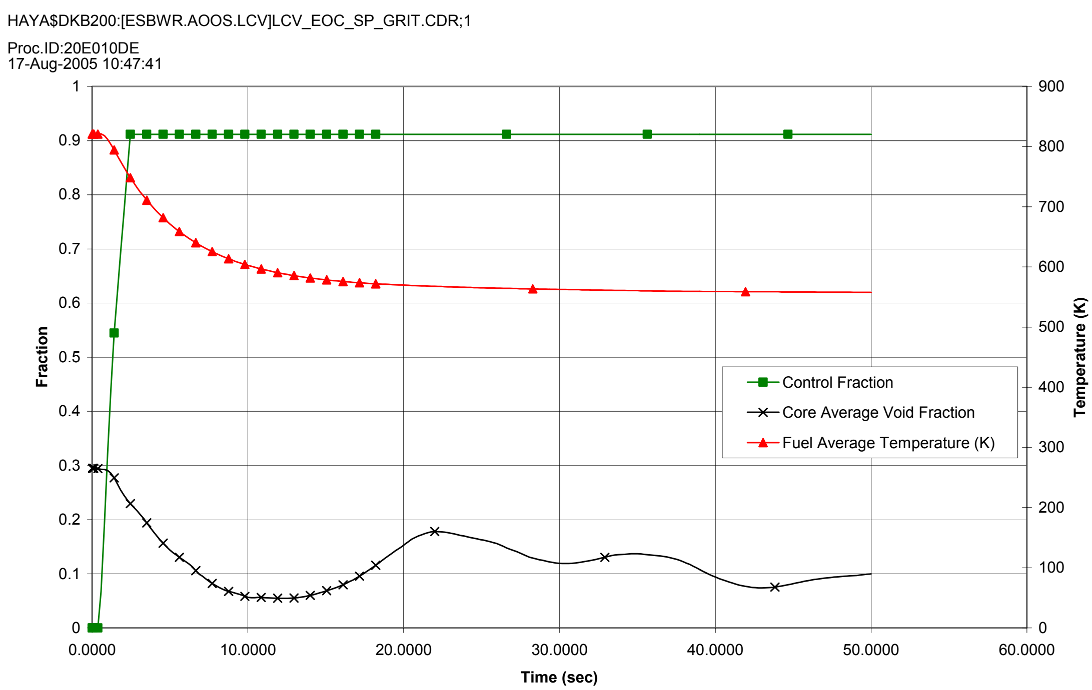


Figure 15.2-10f. Loss of Condenser Vacuum

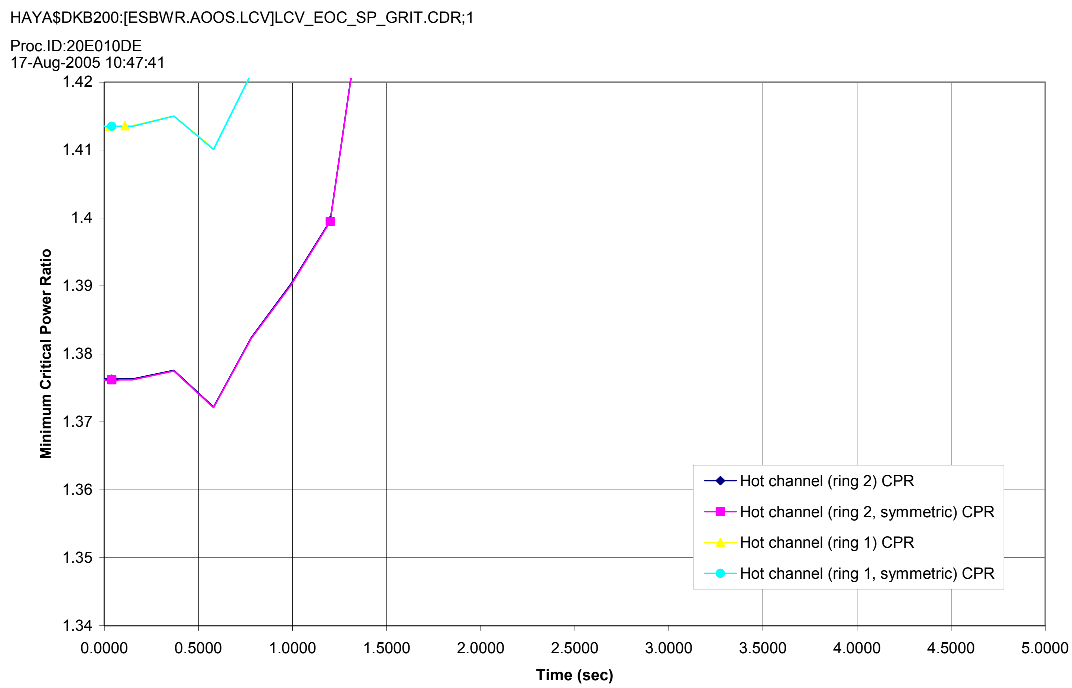


Figure 15.2-10g. Loss of Condenser Vacuum

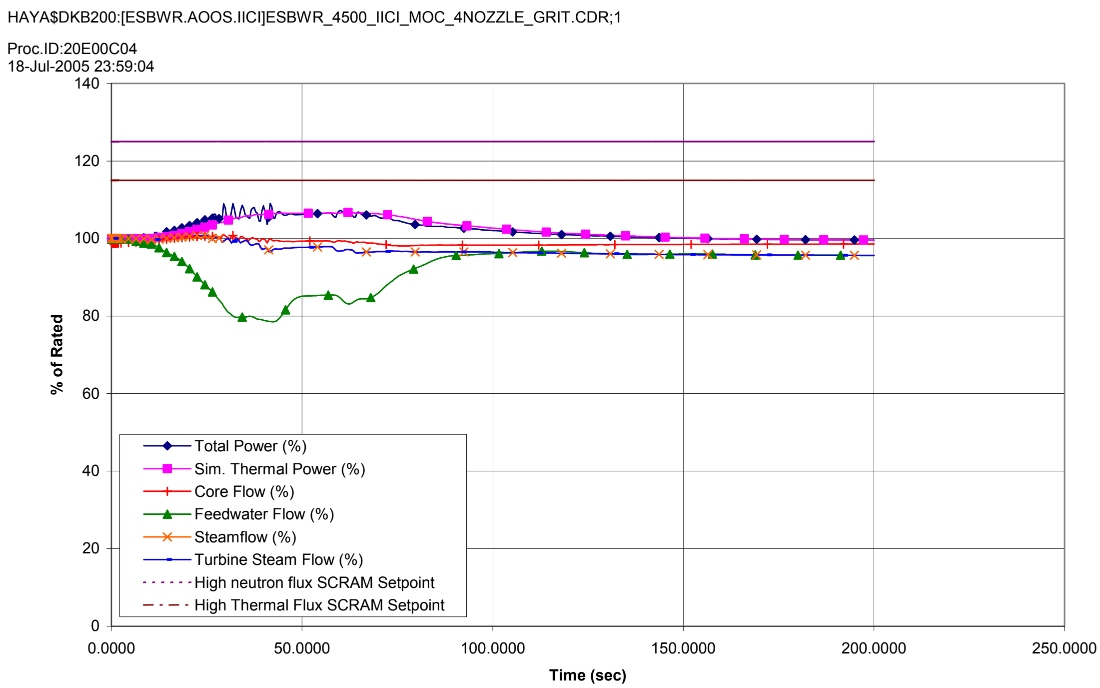


Figure 15.2-11a. Inadvertent Isolation Condenser Initiation

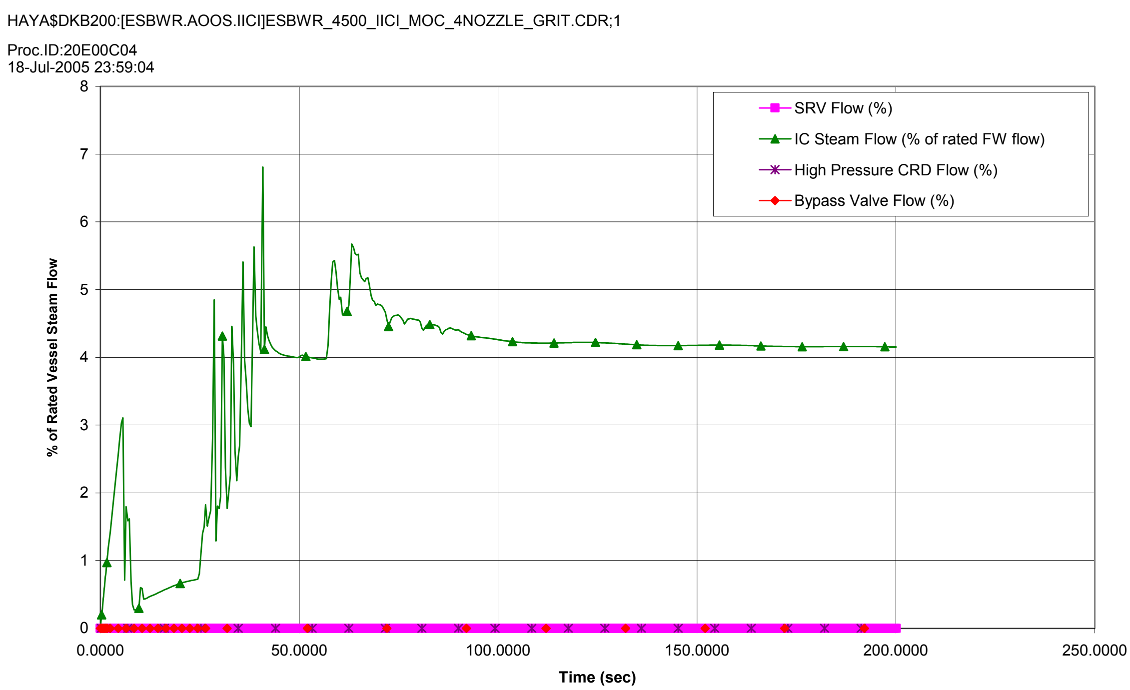


Figure 15.2-11b. Inadvertent Isolation Condenser Initiation

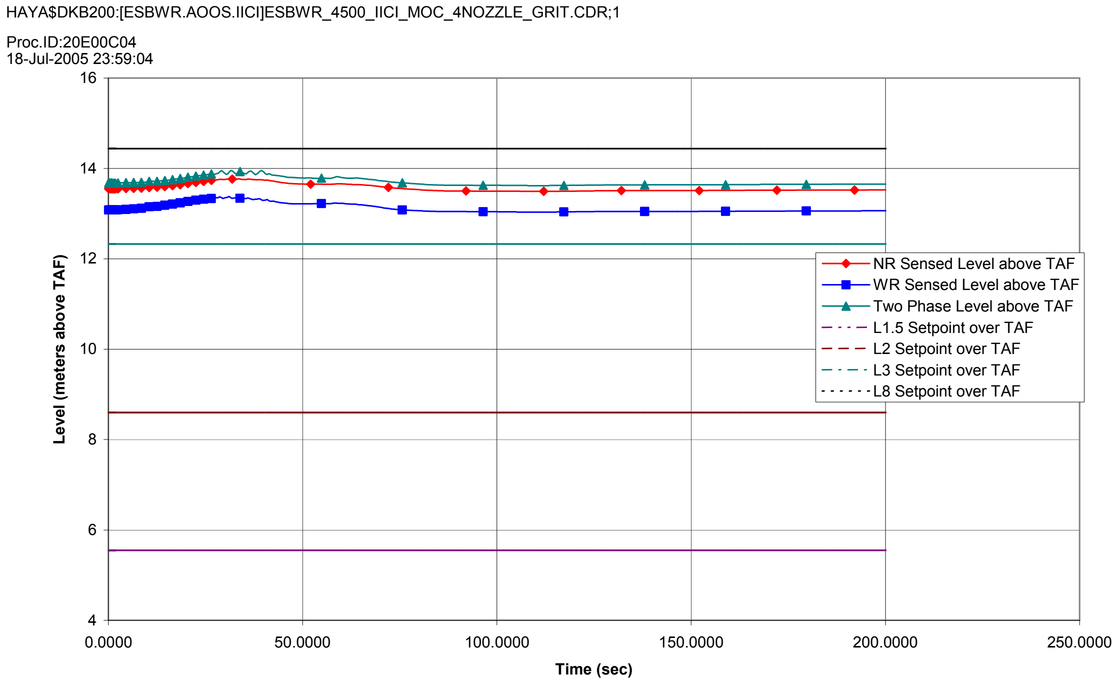


Figure 15.2-11c. Inadvertent Isolation Condenser Initiation

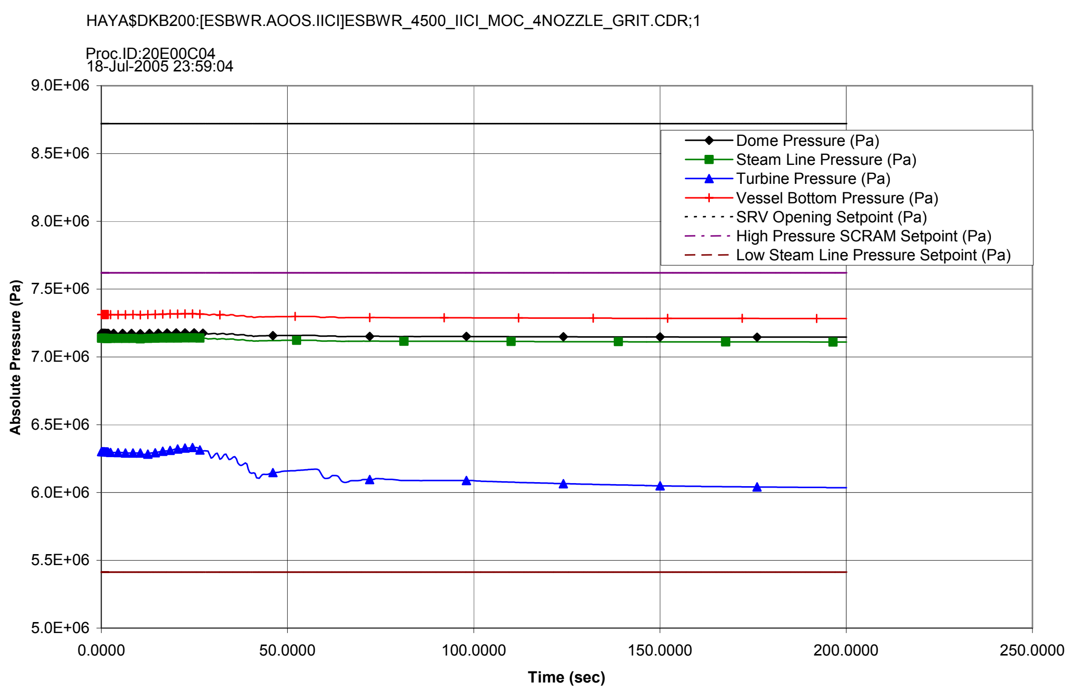


Figure 15.2-11d. Inadvertent Isolation Condenser Initiation

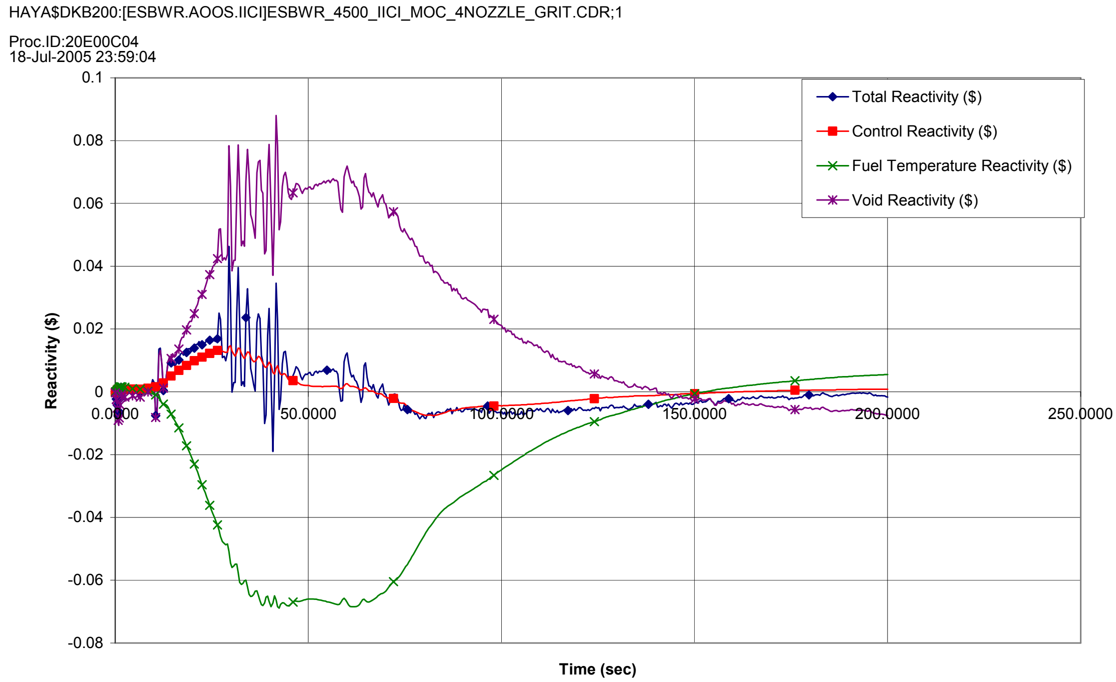


Figure 15.2-11e. Inadvertent Isolation Condenser Initiation

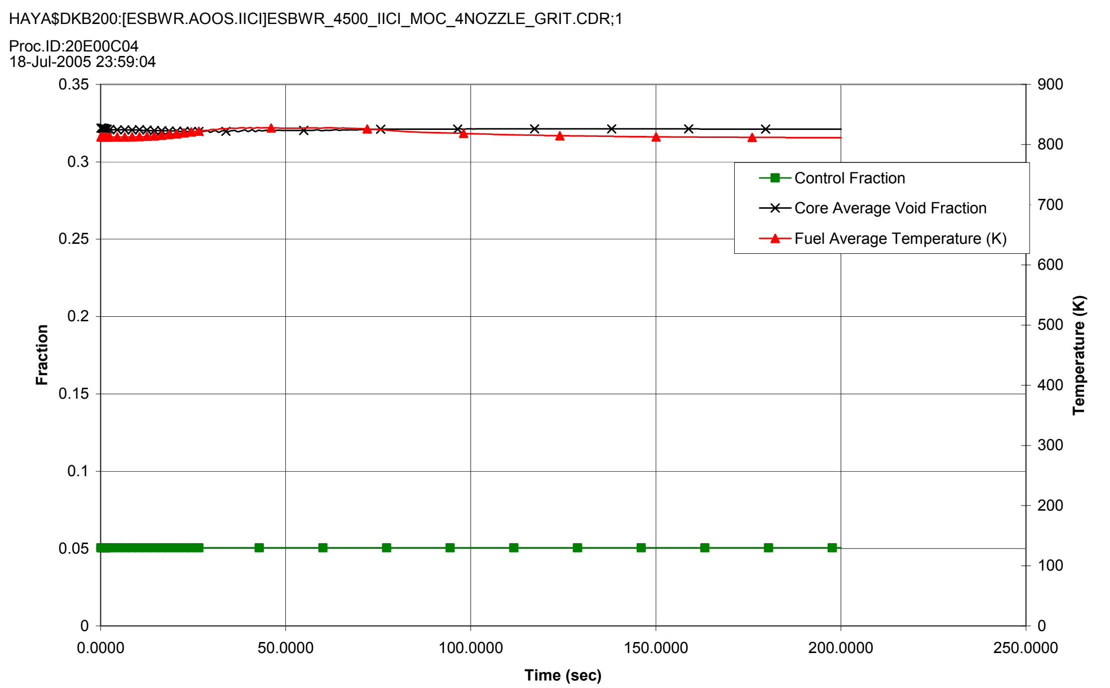


Figure 15.2-11f. Inadvertent Isolation Condenser Initiation

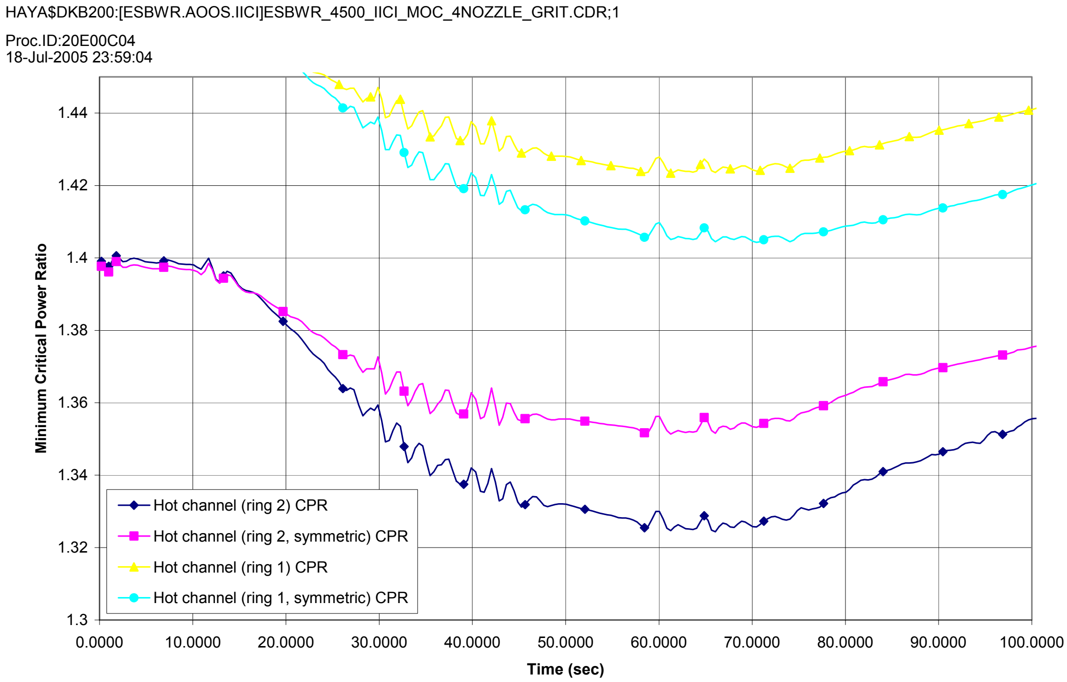


Figure 15.2-11g. Inadvertent Isolation Condenser Initiation

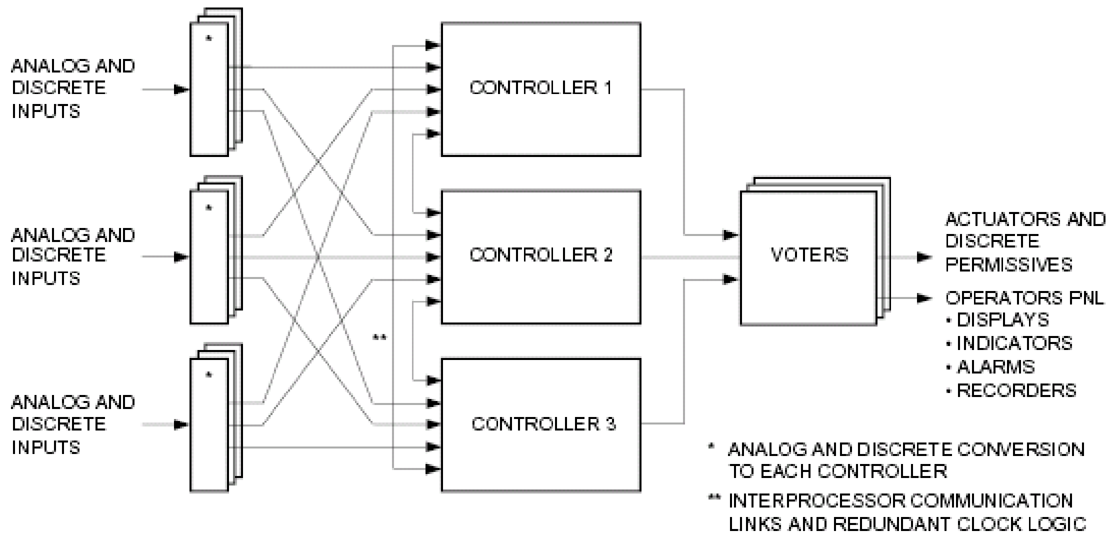


Figure 15.2-12. Simplified Block Diagram of Fault-Tolerant Digital Controller System

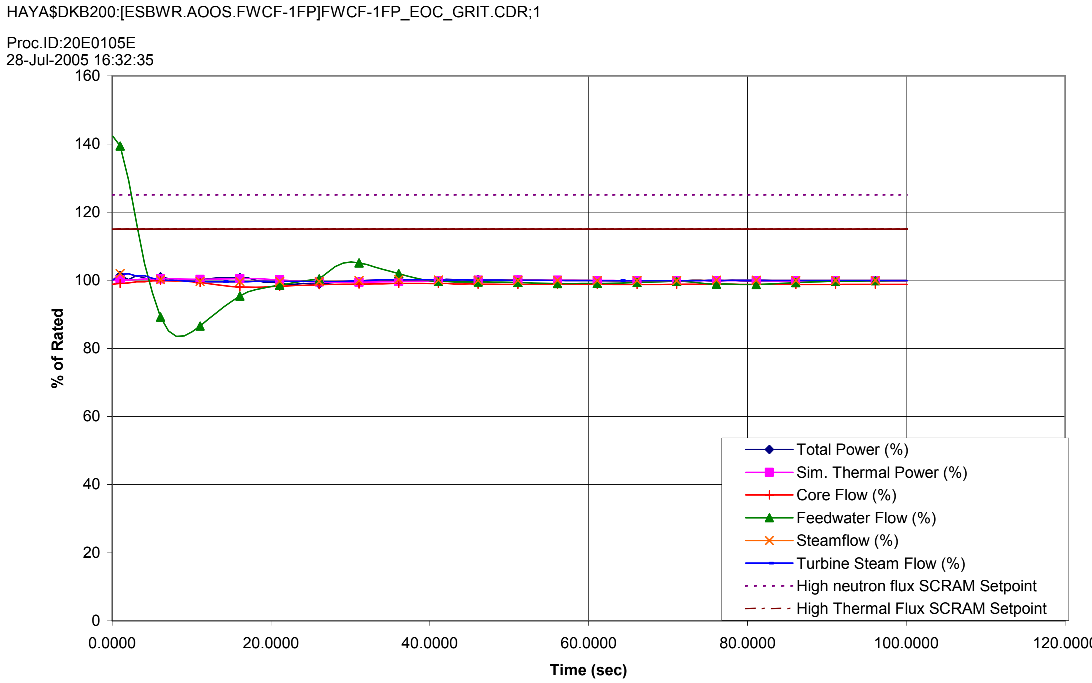


Figure 15.2-13a. Runout of One Feedwater Pump

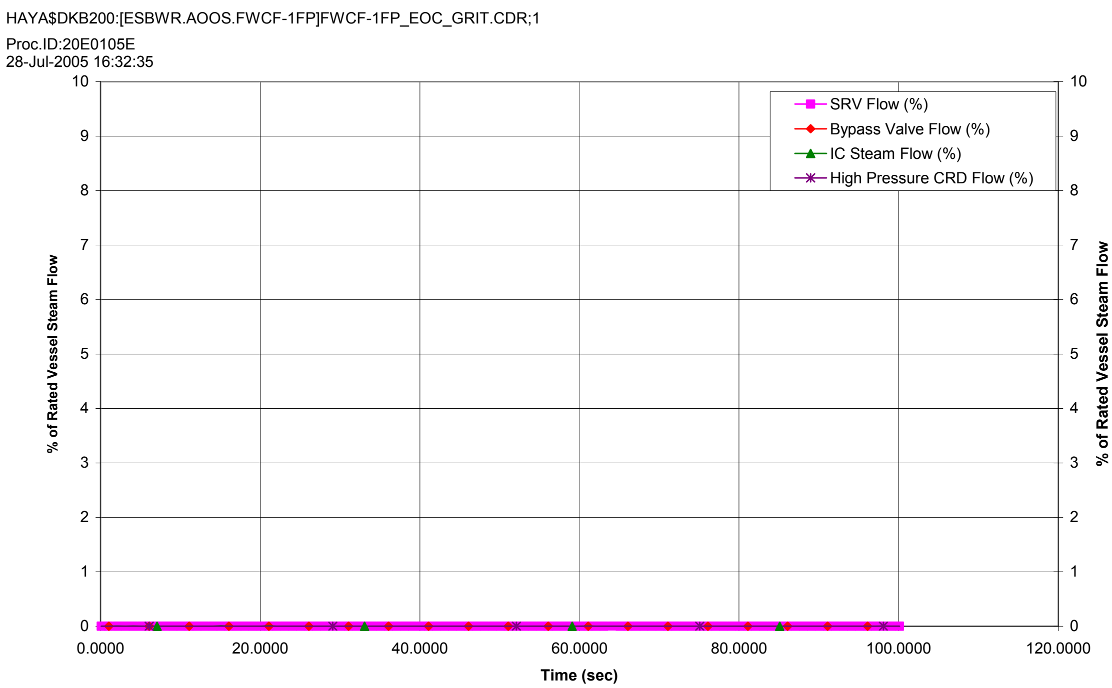


Figure 15.2-13b. Runout of One Feedwater Pump

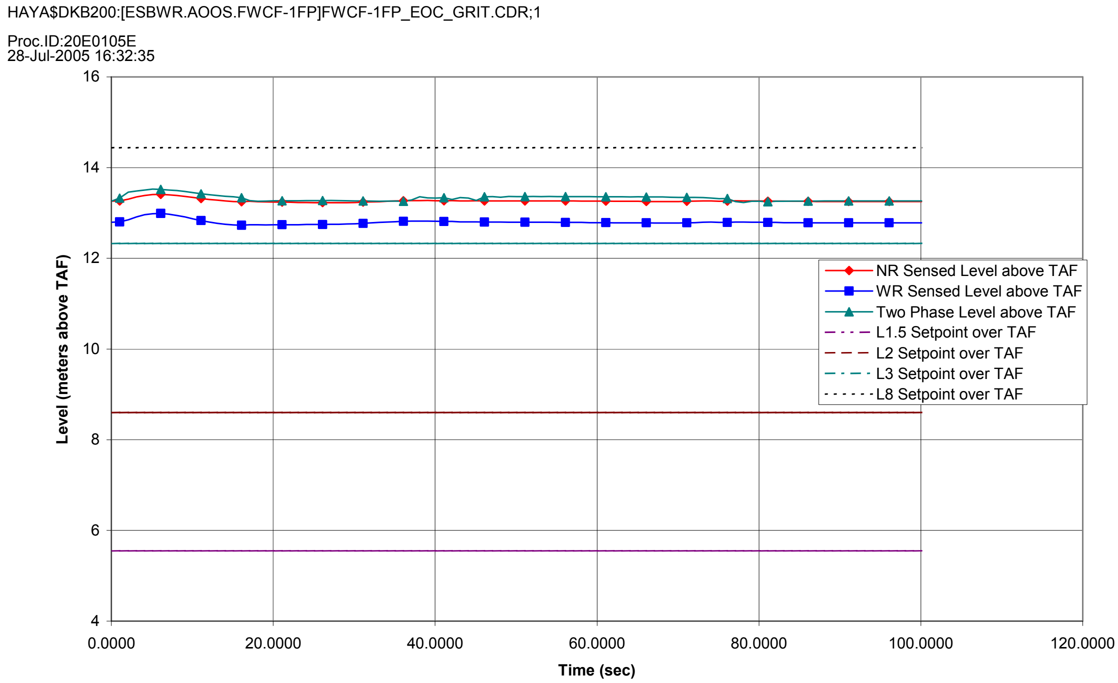


Figure 15.2-13c. Runout of One Feedwater Pump

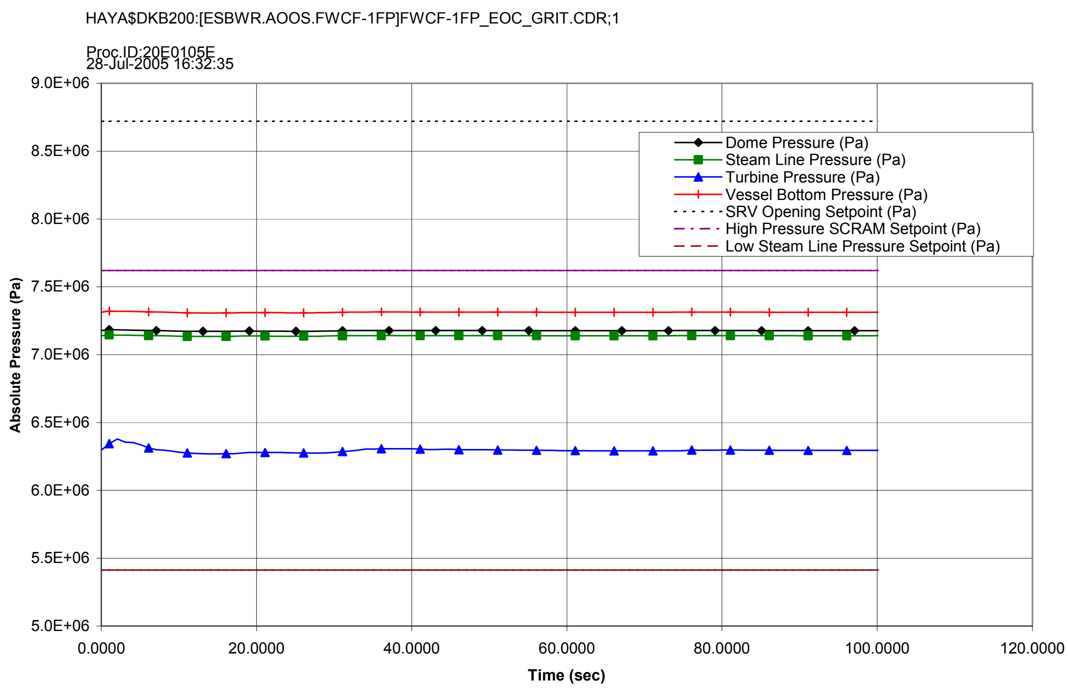


Figure 15.2-13d. Runout of One Feedwater Pump

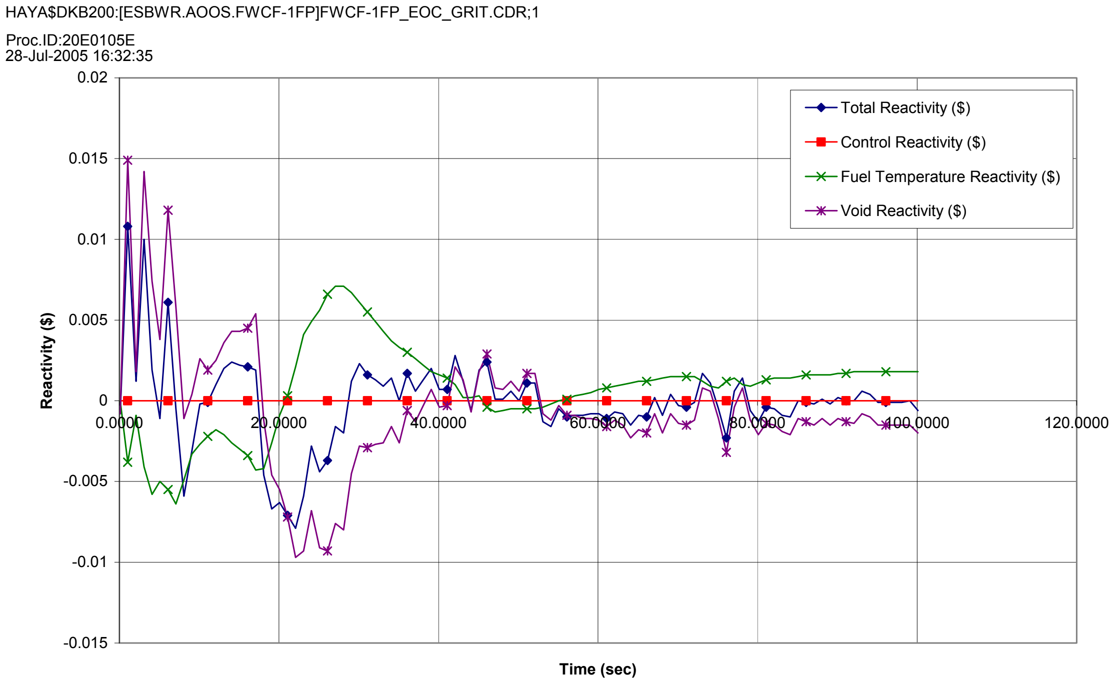


Figure 15.2-13e. Runout of One Feedwater Pump

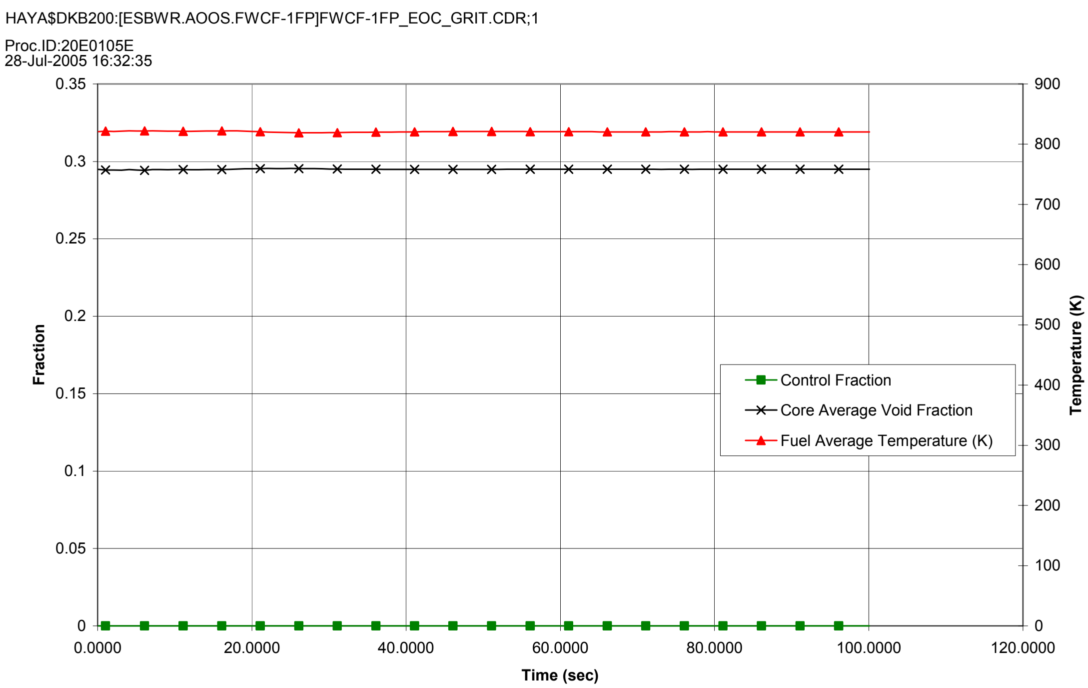


Figure 15.2-13f. Runout of One Feedwater Pump

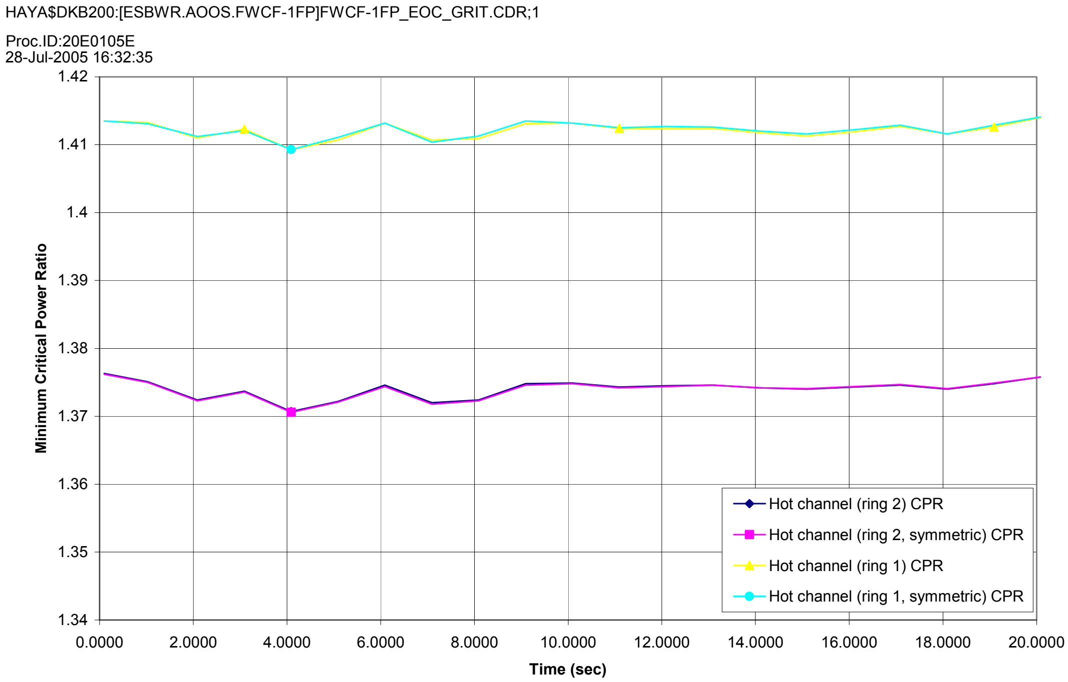


Figure 15.2-13g. Runout of One Feedwater Pump

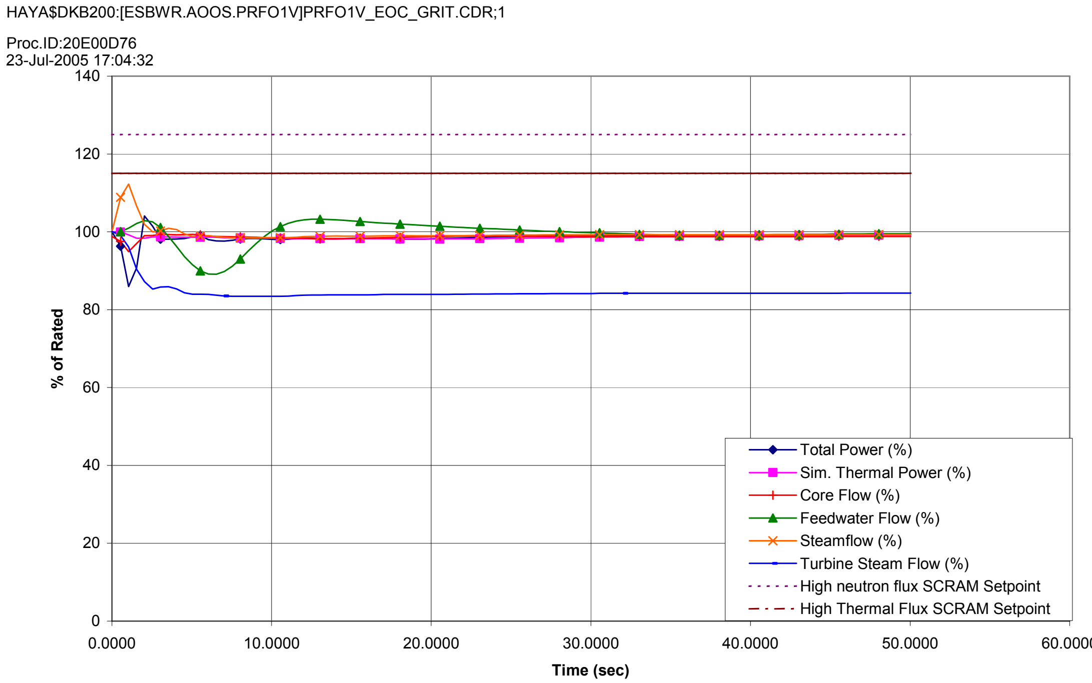


Figure 15.2-14a. Opening of One Turbine Control or Bypass Valve

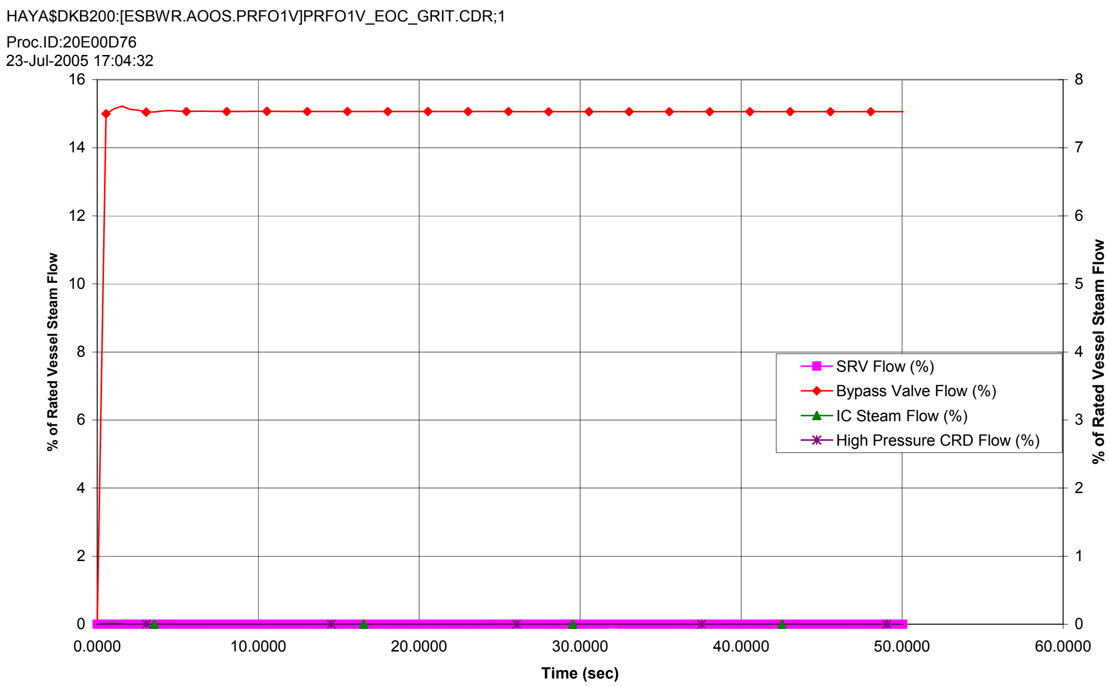


Figure 15.2-14b. Opening of One Turbine Control or Bypass Valve

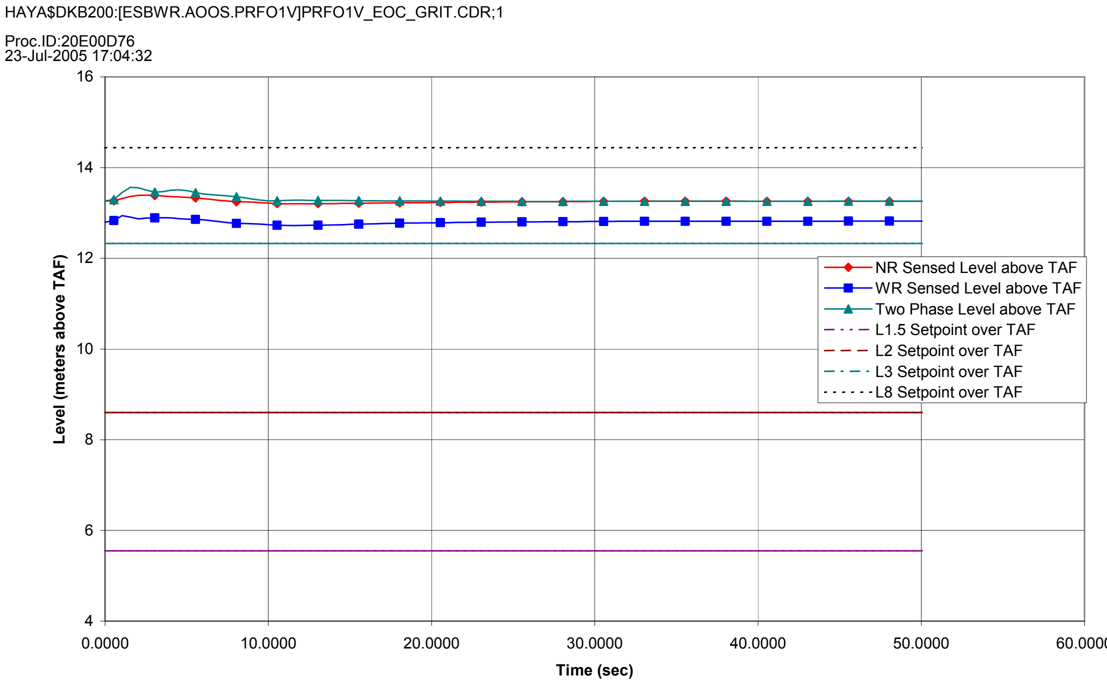


Figure 15.2-14c. Opening of One Turbine Control or Bypass Valve

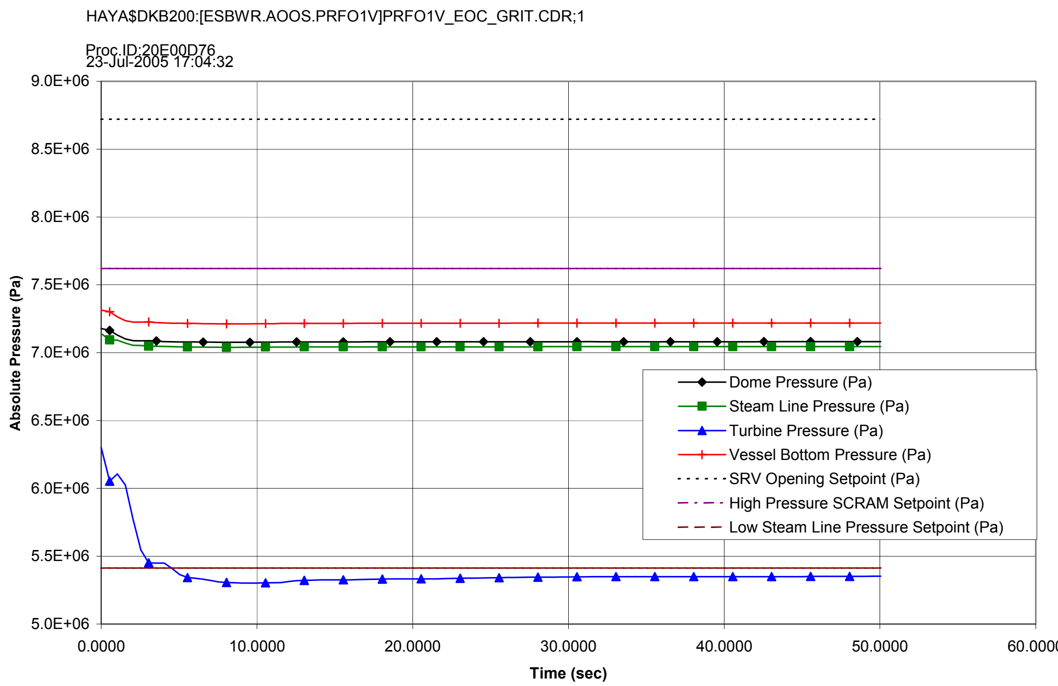


Figure 15.2-14d. Opening of One Turbine Control or Bypass Valve

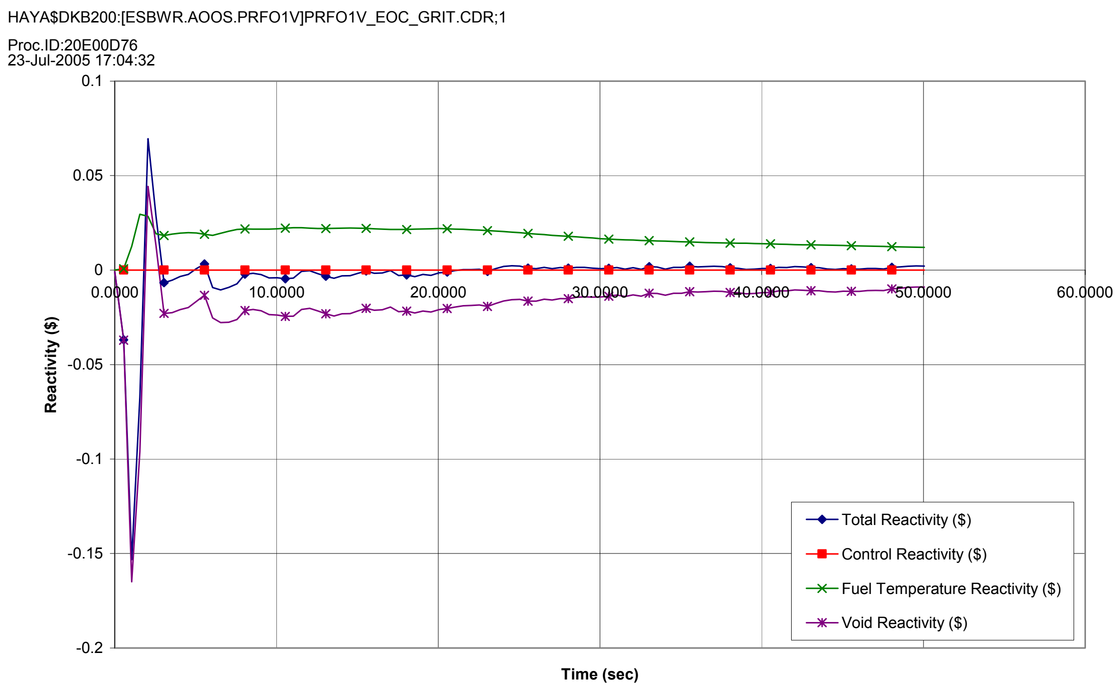


Figure 15.2-14e. Opening of One Turbine Control or Bypass Valve

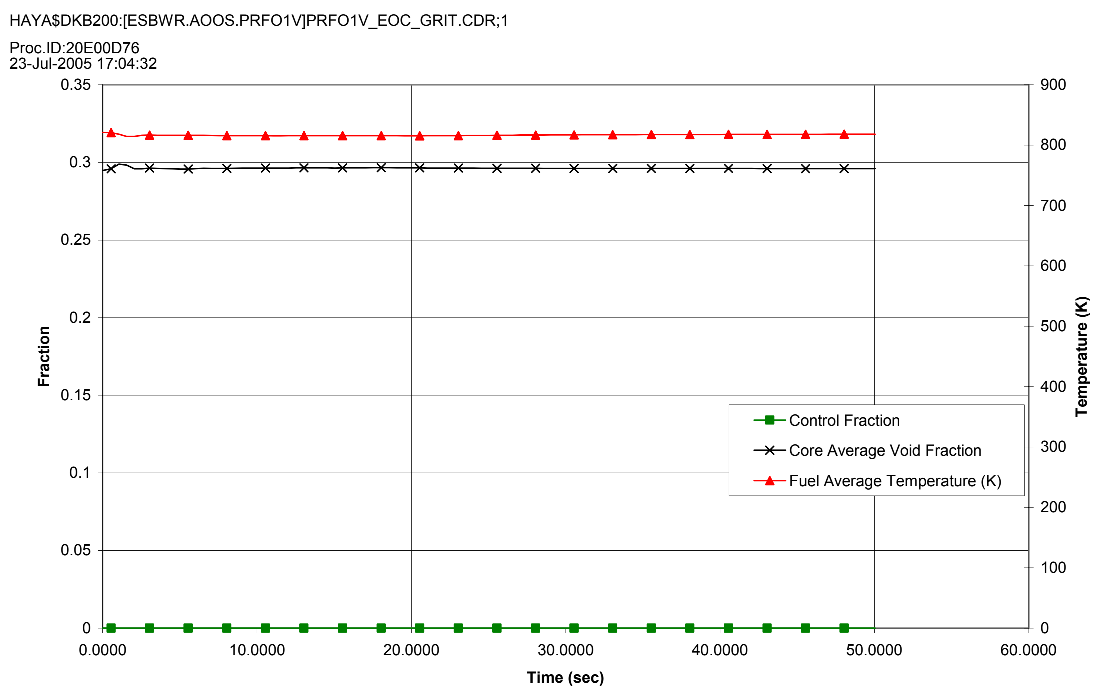


Figure 15.2-14f. Opening of One Turbine Control or Bypass Valve

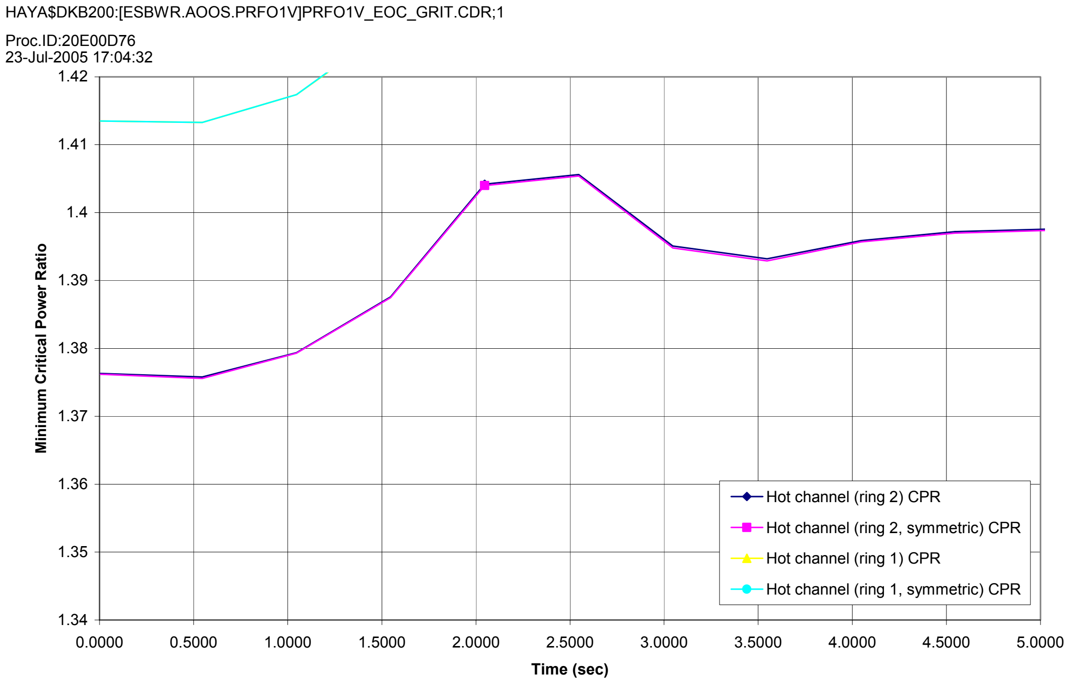


Figure 15.2-14g. Opening of One Turbine Control or Bypass Valve

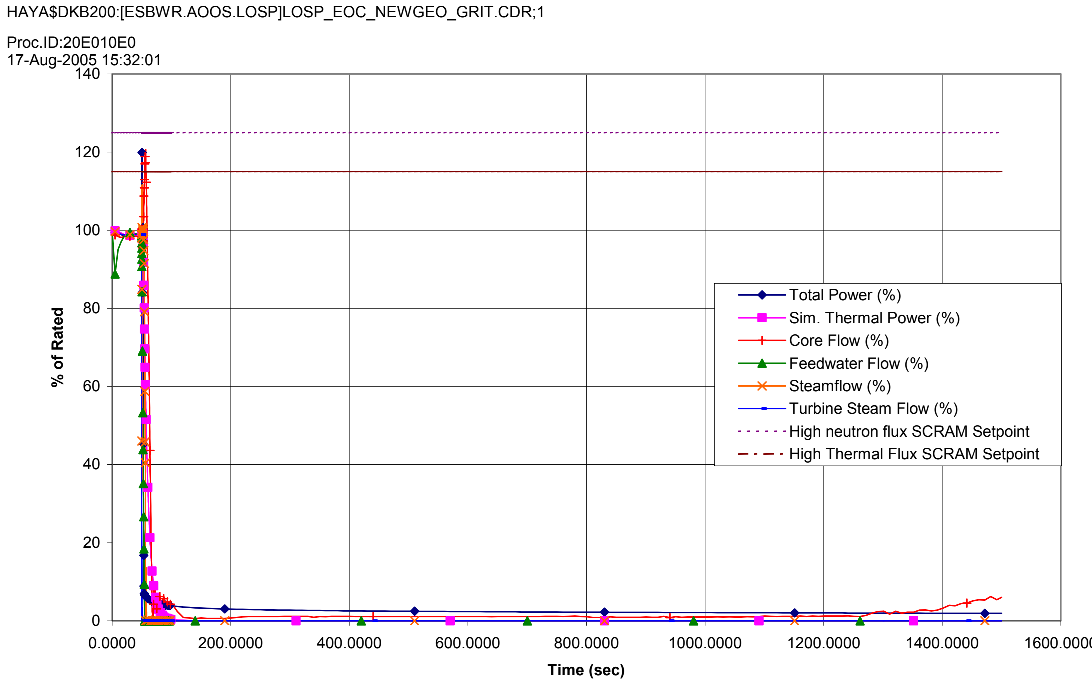


Figure 15.2-15a. Loss of Non-Emergency AC Power to Station Auxiliaries

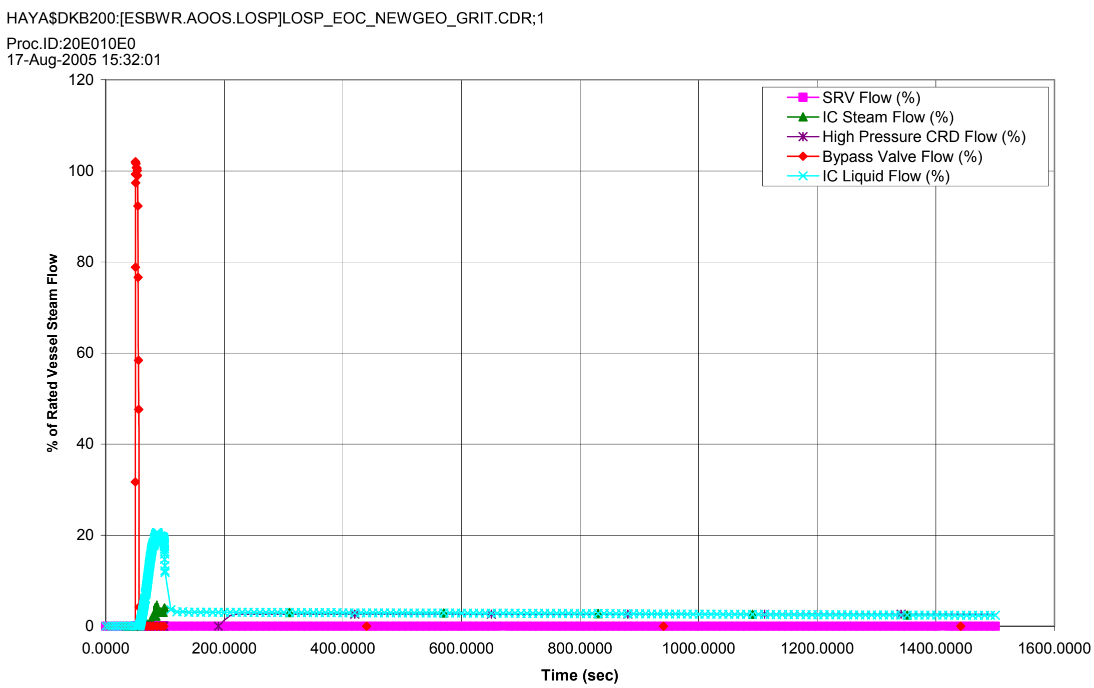


Figure 15.2-15b. Loss of Non-Emergency AC Power to Station Auxiliaries

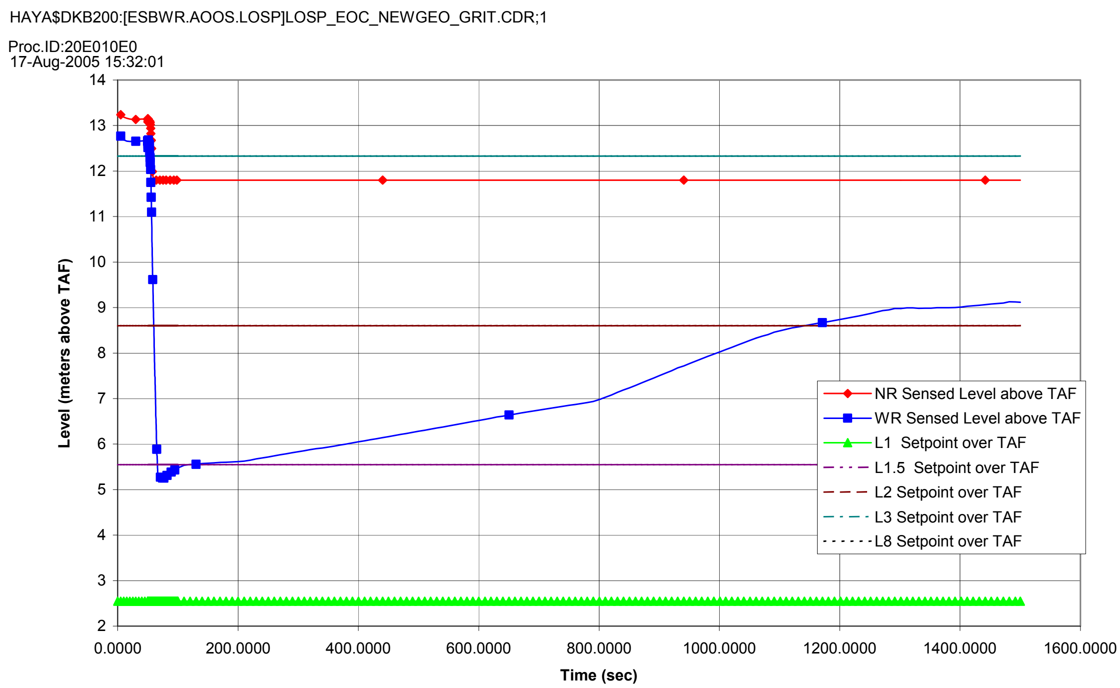


Figure 15.2-15c. Loss of Non-Emergency AC Power to Station Auxiliaries

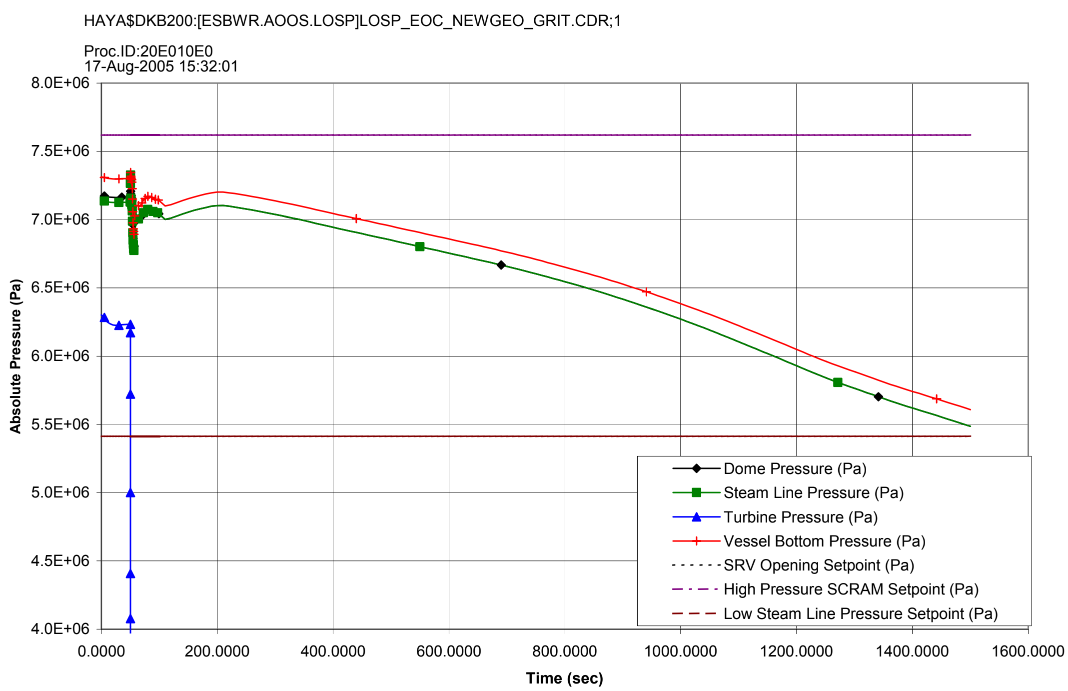


Figure 15.2-15d. Loss of Non-Emergency AC Power to Station Auxiliaries

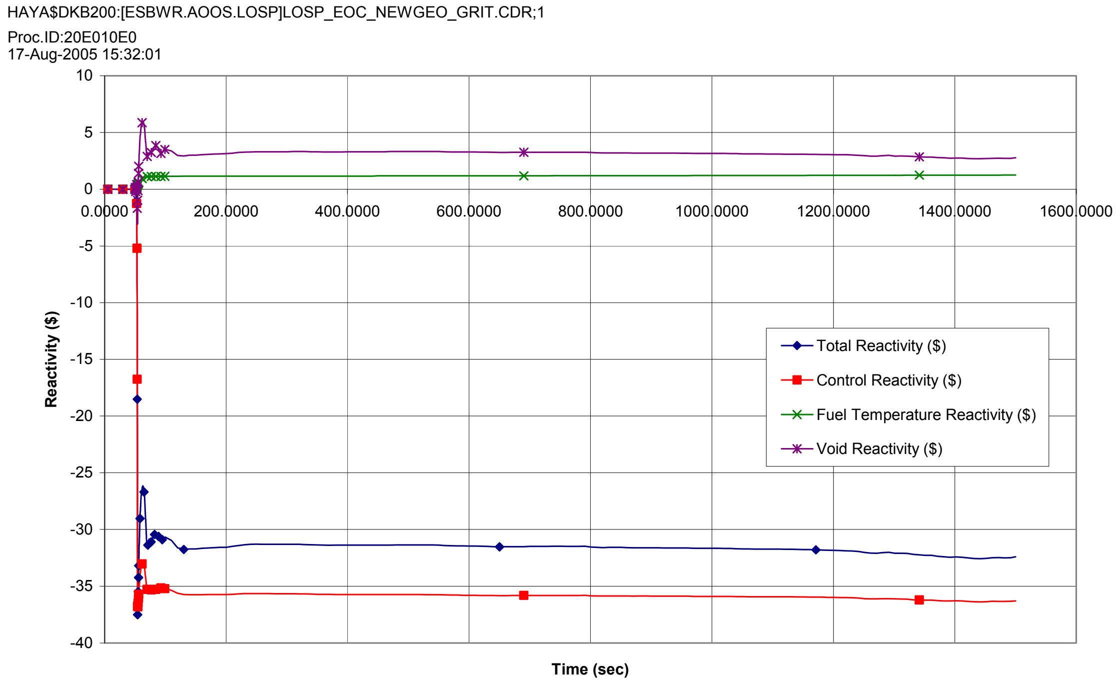


Figure 15.2-15e. Loss of Non-Emergency AC Power to Station Auxiliaries

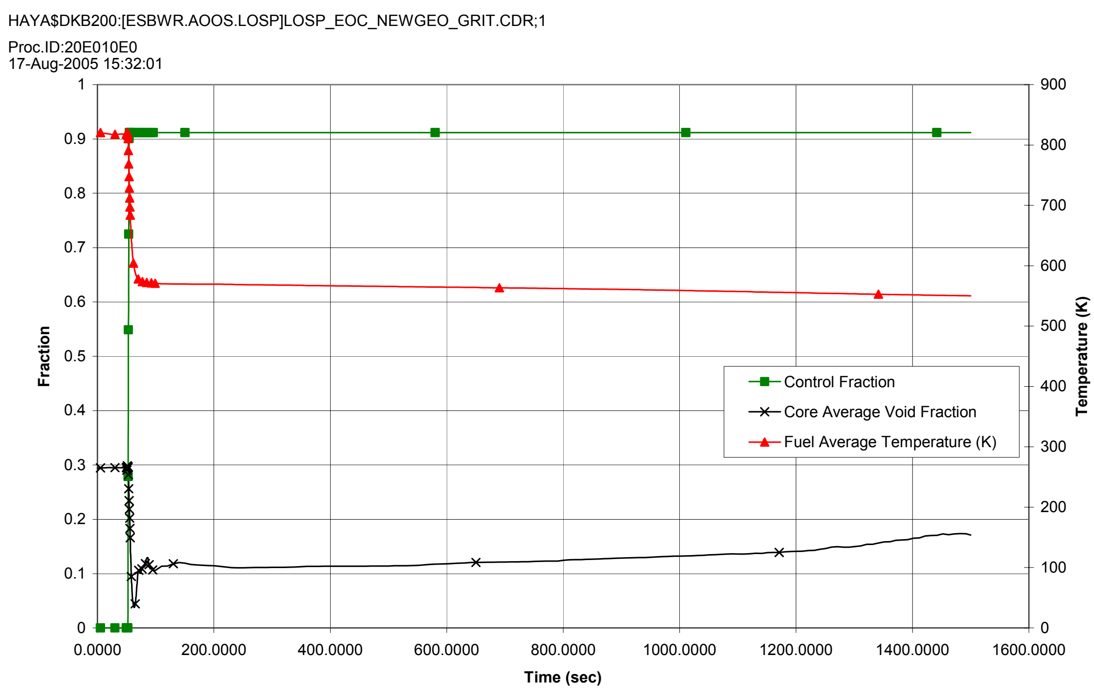


Figure 15.2-15f. Loss of Non-Emergency AC Power to Station Auxiliaries

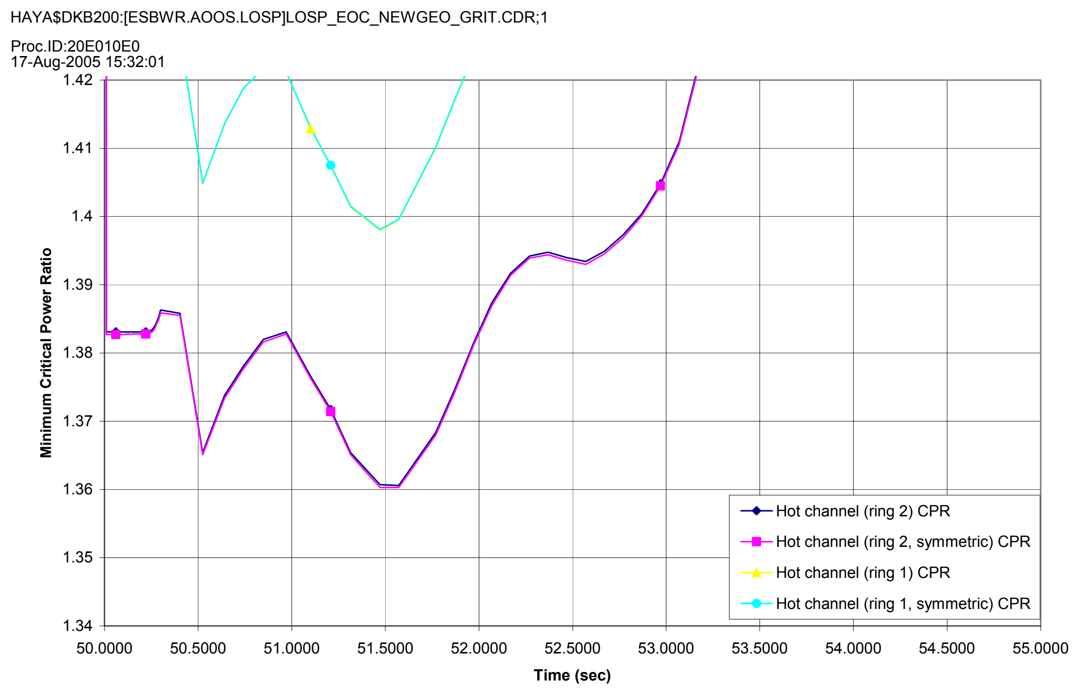


Figure 15.2-15g. Loss of Non-Emergency AC Power to Station Auxiliaries

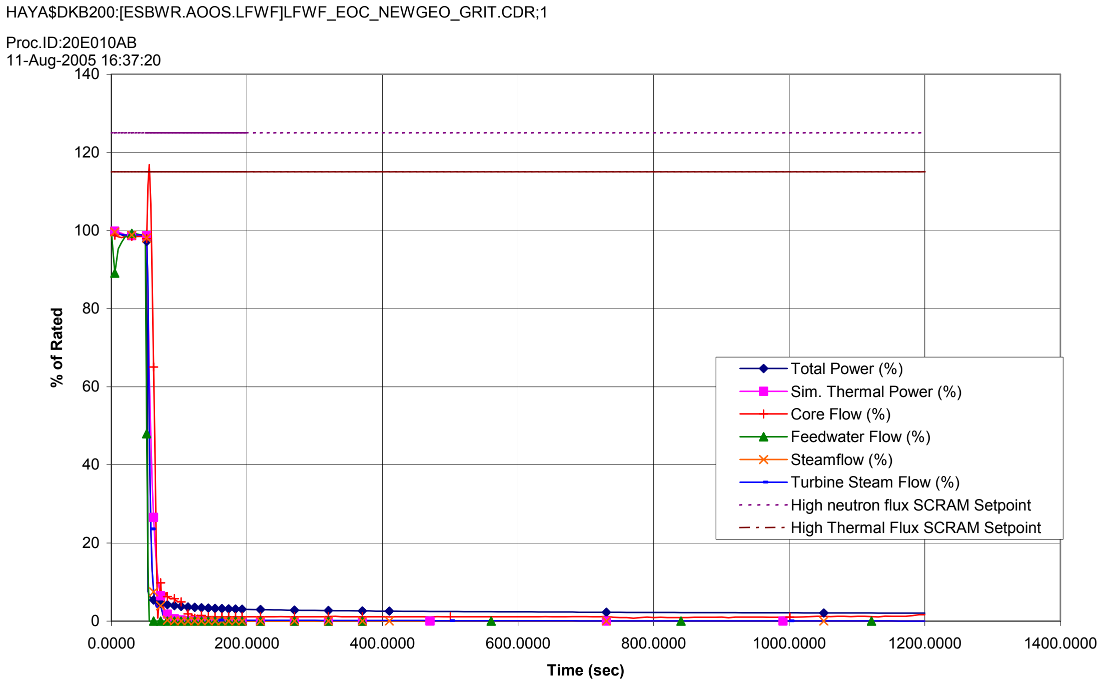


Figure 15.2-16a. Loss of All Feedwater Flow

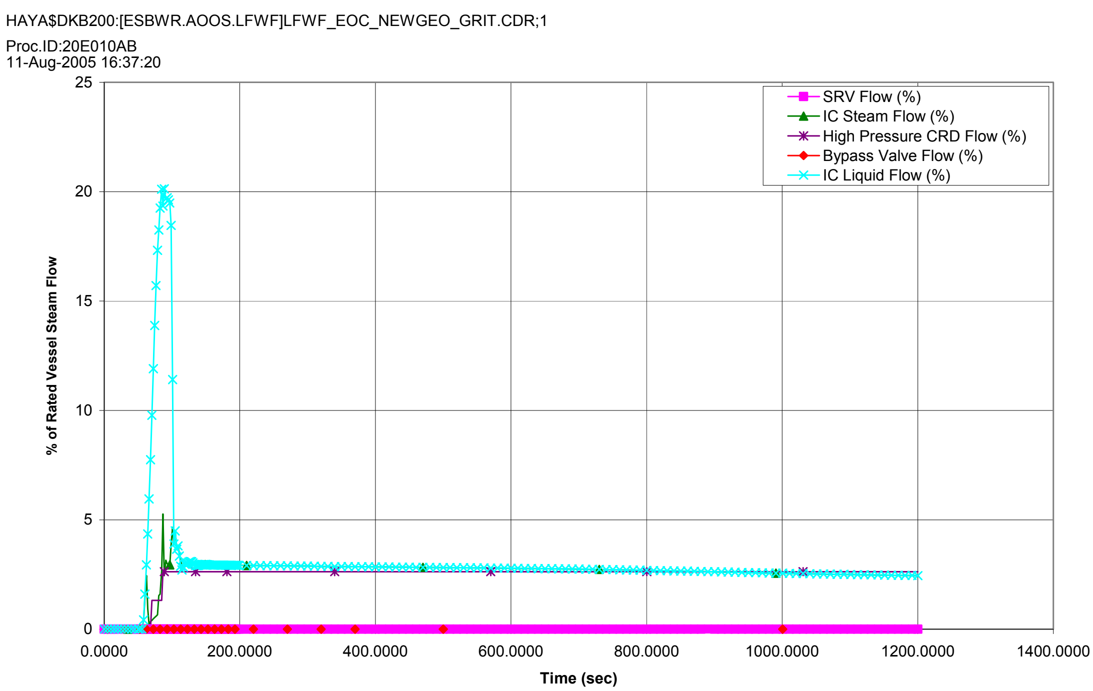


Figure 15.2-16b. Loss of All Feedwater Flow

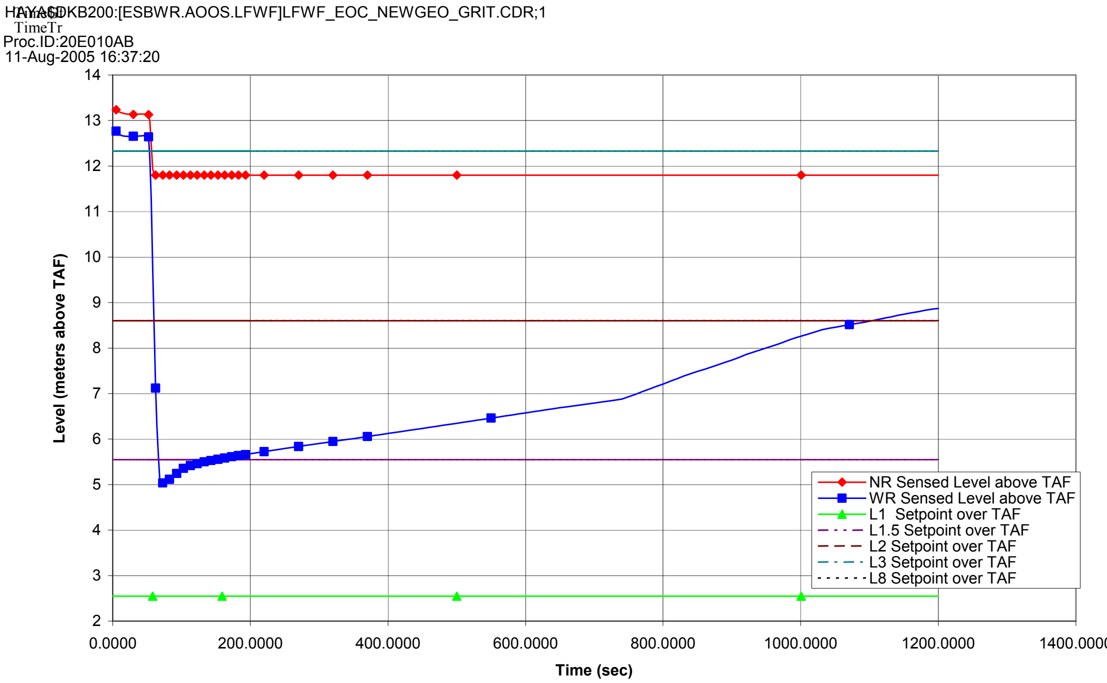


Figure 15.2-16c. Loss of All Feedwater Flow

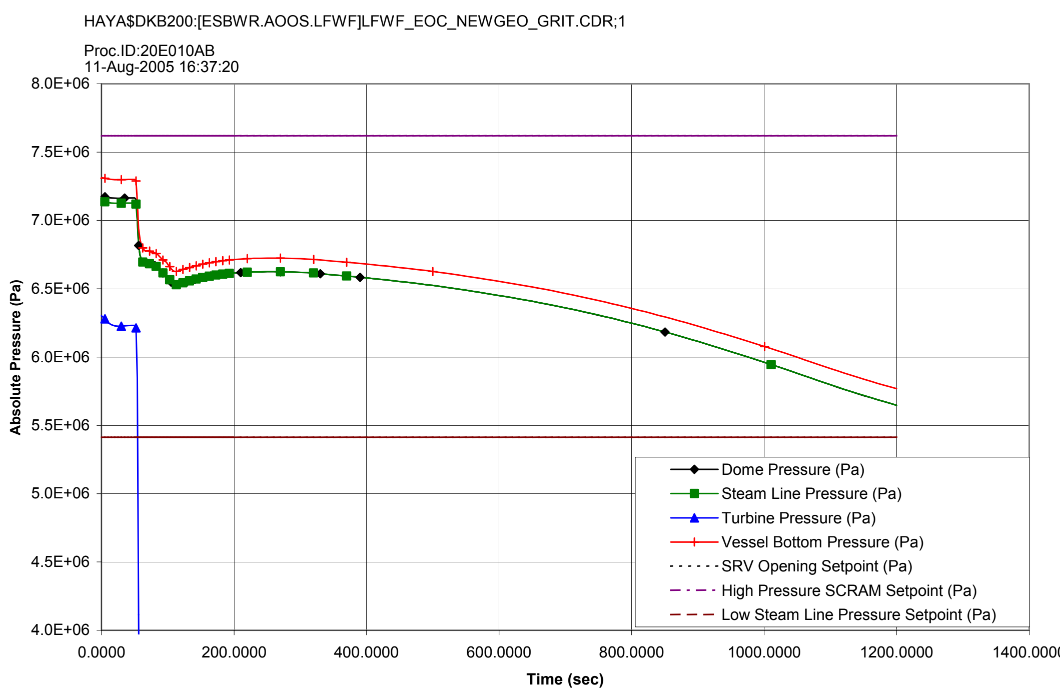


Figure 15.2-16d. Loss of All Feedwater Flow

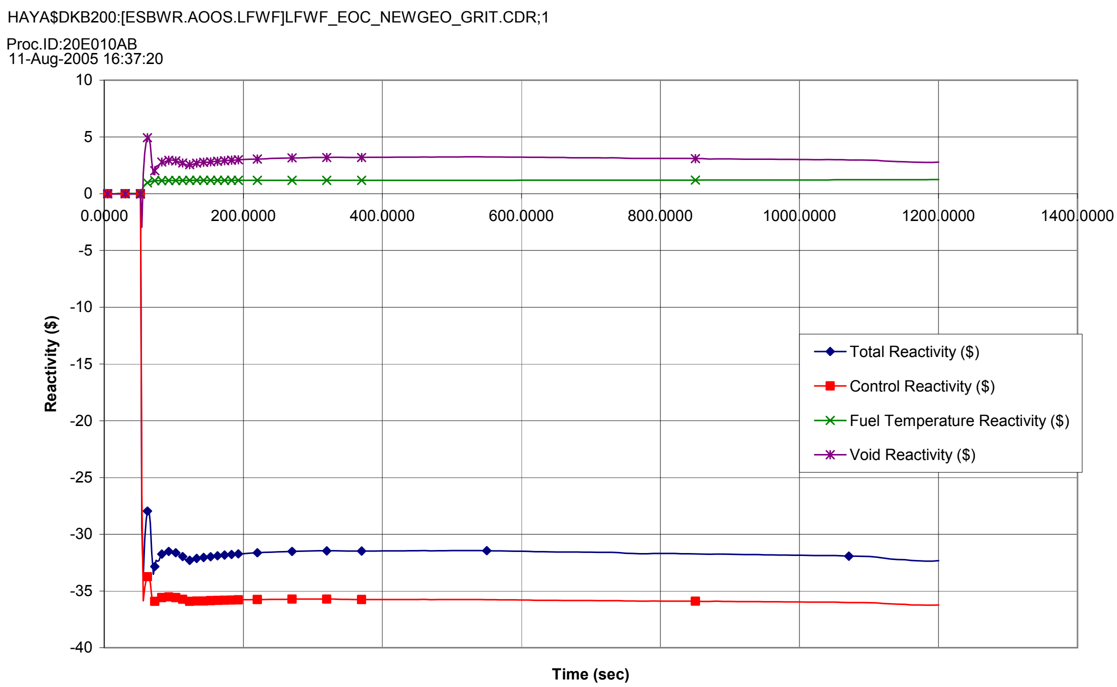


Figure 15.2-16e. Loss of All Feedwater Flow

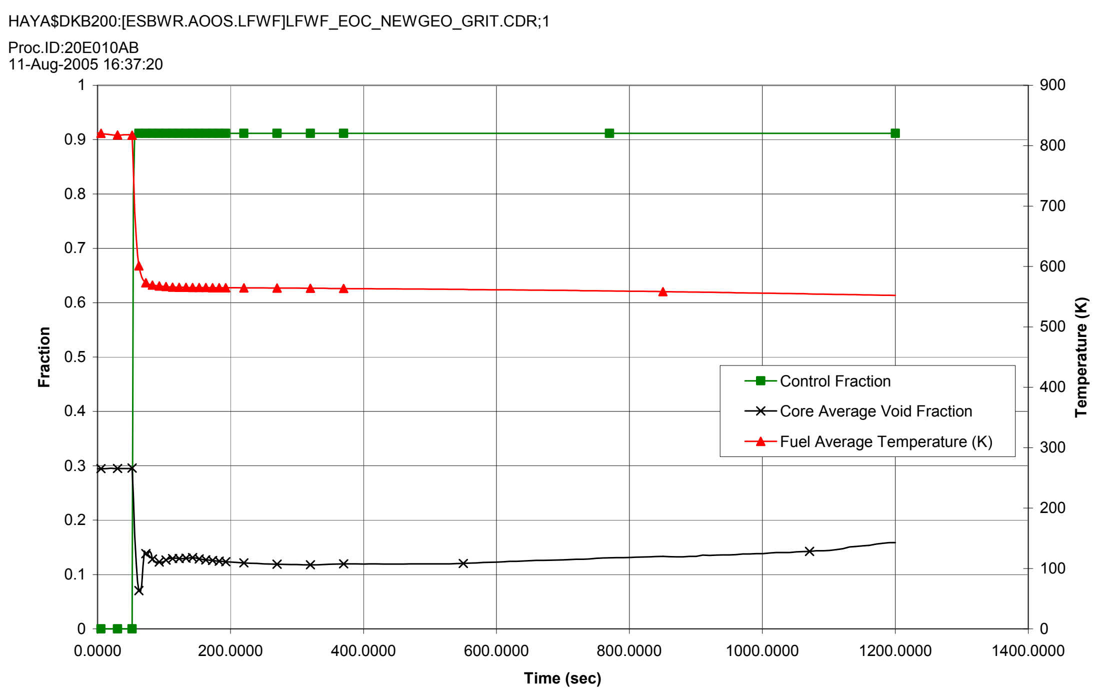


Figure 15.2-16f. Loss of All Feedwater Flow

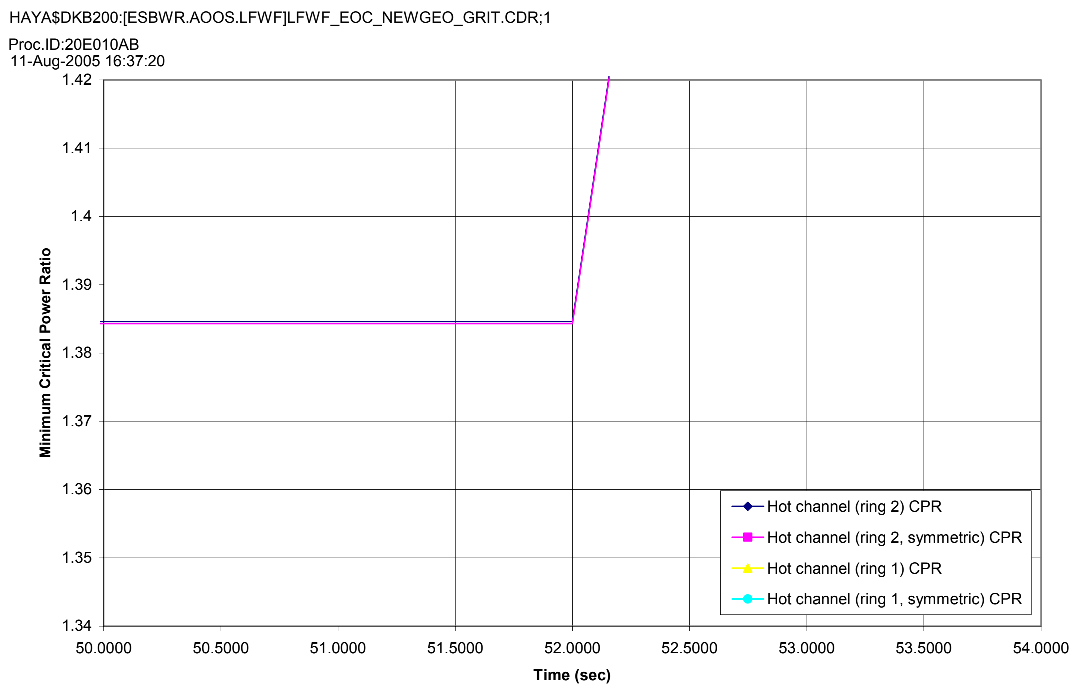


Figure 15.2-16g. Loss of All Feedwater Flow

15.3 ANALYSIS OF INFREQUENT EVENTS

15.3.1 Loss Of Feedwater Heating With Failure of Selected Control Rod Run-In

15.3.1.1 Identification of Causes

A feedwater (FW) heater can be lost in at least two ways:

- steam extraction line to heater is closed; or
- FW is bypassed around heater.

The first case produces a gradual cooling of the FW. In the second case, the FW bypasses the heater and no heating of the FW occurs. In either case, the reactor vessel receives colder FW. The maximum number of FW heaters that can be tripped or bypassed by a single event represents the most severe event for analysis considerations.

The ESBWR is designed such that no single operator error or equipment failure shall cause a loss of more than 55.6°C (100°F) FW heating. The reference steam and power conversion system shown in Section 10.1 meets this requirement. In fact, the FW temperature drop based on the reference heat balance shown in Section 10.1 is less than 39°C (70°F). Therefore, the analyzed FW temperature drop shown in Table 15.2-1 is conservative.

This event conservatively assumes the loss of FW heating shown in Table 15.2-1, which causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient. However, the power increase is slow.

The Feedwater Control System (FWCS) logic is provided in Subsection 7.7.3, and includes logic provided to mitigate the effects of a loss of FW heating capability. The system is constantly monitoring the actual FW temperature and comparing it with a reference temperature. When a loss of FW heating is detected [i.e., when the difference between the actual and reference temperatures exceeds a ΔT setpoint], the FWCS sends an alarm to the operator and sends a signal to the Rod Control and Information System (RC&IS) to initiate the selected control rods run-in (SCRRI) function to automatically reduce the reactor power and avoid a scram. However, for this event, SCRRI is assumed to fail and reactor scram on high simulated thermal power is not credited due to uncertainties. Therefore a new steady state is reached

15.3.1.2 Sequence of Events and Systems Operation

Sequence of Events

Table 15.3-2 lists the sequence of events for Figure 15.3-1.

Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection and reactor protection systems.

The high simulated thermal power trip (STPT) scram is the primary protection system trip in mitigating the effects of this event. However, credit was not taken for this scram to consider the possibility that, for a similar case with a somewhat lower loss of heating, the scram setpoint might not be reached, while the consequences would only be slightly less severe for this case than the event analyzed here.

15.3.1.3 Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by programming a change in FW enthalpy corresponding to the assumed loss in FW heating, shown in Table 15.2-1.

Results

Reactor scram should be initiated during this event. However, as explained above, credit for STPT scram was not taken. The nuclear system pressure does not significantly change [< 0.02 MPa (3.0 psi)] during the event, and consequently, the RCPB is not threatened.

15.3.1.4 Barrier Performance

As noted previously, the effects of this event do not result in any temperature or pressure transient in excess of the criteria for which the pressure vessel or containment are designed. Therefore, these barriers maintain their integrity and function as designed. In this event, the number of fuel rods that enter transition boiling is bounded by 1000 rods. It was assumed that all rods entering transition boiling fail.

15.3.1.5 Radiological Consequences

The off site dose for this event is less than 2.5 REM Total Effective Dose Equivalent (TEDE) assuming the bounding number (1000 rods) of fuel

15.3.2 Feedwater Controller Failure – Maximum Demand

15.3.2.1 Identification of Causes

See Subsection 15.2.4.2. This event assumes multiple control system failures, to simultaneously increase the flow in multiple FW pumps to their maximum limit.

15.3.2.2 Sequence of Events and Systems Operation

Sequence of Events

With excess FW flow, the water level rises to the high water level reference point (Level 8), at which time the FW pumps are run back, the main turbine is tripped and a scram is initiated. Table 15.3-3 lists the sequence of events for Figure 15.3-2. The figure shows the changes in important variables during this event.

Because Level 8 is located near the top of the separators, some moisture entrainment and carry-over to the turbine and bypass valve may occur. While this is potentially harmful to the turbine's integrity, it has no safety implications for the plant.

Identification of Operator Actions

The operator should:

- Follow the scram procedure.

- Observe that FW flow runback due to high water level has terminated the failure event.
- Switch the FW controller from auto to manual control to try to regain a correct output signal.
- Identify causes of the failure and report all key plant parameters during the event.

15.3.2.2.1 System Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems. Important system operational actions for this event are tripping of the main turbine, FW flow runback, and scram due to high water level (Level 8).

15.3.2.3 Core and System Performance

15.3.2.3.1 Input Parameters and Initial Conditions

The total FW flow for all pumps runout is provided in Table 15.2-1.

15.3.2.3.2 Results

The simulated runout of all FW pumps is shown in Figure 15.3-2. The high water level turbine trip and FW pump runback are initiated early in the event as shown in Table 15.3-3. Scram occurs and limits the neutron flux peak and fuel thermal transient so that no fuel damage occurs. The Turbine Bypass System opens to limit peak pressure in the steamline near the SRVs and the peak pressure at the bottom of the vessel. The peak absolute pressure in the bottom of the vessel remains below the ASME code upset limit. Peak steam line absolute pressure near the SRVs remains below the setpoint of the SRVs.

The water level gradually drops, and can reach the Low Level reference point (Level 2), activating the IC system for long-term level control and the HP_CRD system to permit a slow recovery to the desired level

The COL applicant will provide reanalysis of this event for the specific initial core configuration.

15.3.2.4 Barrier Performance

As previously noted, the effect of this event does not result in any temperature or pressure transient in excess of the criteria for which the pressure vessel or containment are designed. Therefore, these barriers maintain their integrity and function as designed. In this event, there are no fuel rods that enter transition boiling.

15.3.2.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event

15.3.3 Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

15.3.3.1 Identification of Causes

The ESBWR Steam Bypass and Pressure Control (SB&PC) system uses a triplicated digital control system. The SB&PC system controls the turbine control valves and turbine bypass valves to maintain reactor pressure. As presented in Subsection 15.2.4.2, no credible single failure in the control system results in a maximum demand to all actuators for all turbine control valves and bypass valves. A voter or actuator failure may result in an inadvertent opening of one turbine control valve or one turbine bypass valve. In this case, the SB&PC system senses the pressure change and commands the remaining control valves to close, and thereby automatically mitigates the transient and maintains reactor power and pressure.

As presented in Subsection 15.2.4.2, multiple failures might cause the SB&PC system to erroneously issue a maximum demand to all turbine control valves and bypass valves. Should this occur, all turbine control valves and bypass valves could be fully opened. However, the probability of this event is extremely low, and thus, the event is considered as a limiting fault.

15.3.3.2 Sequence of Events and Systems Operation

15.3.3.2.1 Sequence of Events

Table 15.3-4 lists the sequence of events for Figure 15.3-3.

15.3.3.2.2 Identification of Operator Actions

If the reactor scrams as a result of the isolation caused by the low pressure at the turbine inlet in the run mode, the following sequence of operator actions is expected during the course of the event.

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure
- Monitor reactor water level and pressure;
- Monitor turbine coastdown and break vacuum before the loss of steam seals (check turbine auxiliaries for proper operation);
- Observe that ICS is initiated on low-water level or MSIV closure;
- Use ICS to control reactor pressure and level;
- Cooldown the reactor per standard procedure if a restart is not intended; and

Complete the scram report and initiate a maintenance survey of the SB&PC system before reactor restart.

15.3.3.2.3 Systems Operations

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems, except as otherwise noted.

15.3.3.3 Core and System Performance

15.3.3.3.1 Input Parameters and Initial Condition

A five-second isolation valve closure (maximum isolation valve closing time plus instrument delay) instead of a three second closure is assumed when the turbine pressure decreases below the turbine inlet low-pressure setpoint for main steamline isolation initiation. This is within the specification limits of the valve and represents a conservative assumption.

15.3.3.3.2 Results

Figure 15.3-3 presents graphically how the low steam line pressure trips the isolation valve closure, stops vessel depressurization and produces a normal shutdown of the isolated reactor.

Depressurization results in formation of voids in the reactor coolant and causes a decrease in reactor power almost immediately. Position switches on the isolation valves initiate reactor scram.

The isolation limits the duration and severity of the depressurization so that no significant thermal stresses are imposed on the reactor coolant pressure boundary.

15.3.3.4 Barrier Performance

Barrier performance analyses were not required because the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which fuel, pressure vessel or containment are designed. The peak pressure in the bottom of the vessel remains below its ASME code faulted limit for the RCPB. Peak steam line absolute pressure near the SRVs remains below the setpoint of the SRVs.

15.3.3.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.3.4 Pressure Regulator Failure–Closure of All Turbine Control and Bypass Valves

15.3.4.1 Identification of Causes

The ESBWR Steam Bypass and Pressure Control (SB&PC) system uses a triplicated digital control system, instead of an analog system as was originally supplied in BWR/2 through BWR/6 plants. This system is similar to the one used in the ABWR design. The SB&PC system controls turbine control valves and turbine bypass valves to maintain reactor pressure. As presented in Subsection 15.2.4.2, no credible single failure in the control system results in a minimum demand to all turbine control valves and bypass valves. A voter or actuator failure may result in an inadvertent closure of one turbine control valve or one turbine bypass valve if it is open at the time of failure. In this case, the SB&PC system senses the pressure change and commands the remaining control valves or bypass valves, if needed, to open, and thereby automatically mitigates the transient and tries to maintain reactor power and pressure.

No single failure causes the SB&PC system to erroneously issue a minimum demand to all turbine control valves and bypass valves. However, as discussed in Subsection 15.2.4.2, multiple failures might cause the SB&PC system to fail and erroneously issue a minimum demand. Should this occur, it would cause full closure of turbine control valves as well as inhibition of steam bypass flow and thereby increase reactor power and pressure. When this occurs, reactor scram is initiated when the high reactor flux scram setpoint is reached. The SB&PC system design includes provision to mitigate the effects of this postulated multiple failure event. In the event of a detected failure of two channels of the triplicated control system, a turbine trip is automatically initiated. This event is analyzed here as the simultaneous undetected failure of two control processors, called “pressure regulator downscale failure.” However, the probability of this event to occur is extremely low and hence the event is considered as an Infrequent Event rather than an AOO. A probability analysis for this event is provided in Appendix 15A.

15.3.4.2 Sequence of Events and Systems Operation

15.3.4.2.1 Sequence of Events

Table 15.3-5 lists the sequence of events for Figure 15.3-4.

15.3.4.2.2 Identification of Operator Actions

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure
- Monitor reactor water level and pressure;
- Monitor turbine coastdown and break vacuum before the loss of steam seals (check turbine auxiliaries for proper operation);
- Cool down the reactor per standard procedure if a restart is not intended; and
- Complete the scram report and initiate a maintenance survey of pressure regulator before reactor restart.

15.3.4.2.3 Systems Operation

Except for the failures in the SB&PC system, normal plant instrumentation and controls and plant protection and reactor protection systems are assumed to function normally. Specifically, this event takes credit for high neutron flux scram to shut down the reactor.

The turbine control valves, in their servo mode, have a full stroke closure time, from fully open to fully closed, from 2.5 seconds to 3.5 seconds. The worst case of 2.5 seconds is assumed in the analysis.

15.3.4.3 Core and System Performance

A pressure regulator downscale failure is simulated as shown in Figure 15.3-4.

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. When the sensed neutron flux reaches the high neutron flux scram setpoint, a reactor scram is initiated. The neutron flux and pressure increase is limited by the reactor scram.

15.3.4.4 Barrier Performance

The peak pressure in the bottom of the vessel remains below the ASME code limit for the RCPB. The peak absolute pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. The peak vessel bottom absolute pressure is below its ASME Code faulted pressure limit. In this event, the number of fuel rods that enter transition boiling is bounded by 1000 rods. It was assumed that all rods entering transition boiling fail.

15.3.4.5 Radiological Consequences

The off site dose for this event is less than 2.5 REM Total Effective Dose Equivalent (TEDE) assuming the bounding number (1000 rods) of fuel failures.

15.3.5 Generator Load Rejection With Total Turbine Bypass Failure

15.3.5.1 Identification of Causes

Fast closure of the turbine control valves (TCVs) is initiated whenever electrical grid disturbances occur which result in significant loss of electrical load on the generator. The TCVs are required to close as rapidly as possible to prevent excessive overspeed of the turbine-generator (TG) rotor. Closure of the TCVs causes a sudden reduction in steam flow, that results in an increase in system pressure and reactor shutdown.

After sensing a significant loss of electrical load on the generator, the TCVs are commanded to close rapidly. At the same time, the turbine bypass valves are signaled to open in the "fast" opening mode by the SB&PC system, which uses a triplicated digital controller. As presented in Subsection 15.2.4.2, no single failure can cause all turbine bypass valves to fail to open on demand. The worst single failure can only cause one turbine bypass valve to fail to open on demand.

Frequency Basis: Thorough search of domestic plant operating records have identified three instances of bypass failure during 628 bypass system operations. This gives a probability of bypass failure of 0.0048. Combining the actual frequency of a generator load rejection with the failure rate of bypass, yields a frequency of a generator load rejection with bypass failure of 0.0036 event/plant per year. With the triplicated fault-tolerant design used in ESBWR, this failure frequency is lowered by at least a factor of 100. Therefore, this event is classified as an Infrequent Event rather than an AOO.

15.3.5.2 Sequence of Events and System Operation

15.3.5.2.1 Sequence of Events

A loss of generator electrical load at high power with failure of all bypass valves produces the sequence of events listed in Table 15.3-6.

15.3.5.2.2 Identification of Operator Actions

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure

- Verify proper bypass valve performance;
- Observe that the feedwater/level controls have maintained the reactor water level at a satisfactory value;
- Observe that the pressure regulator is controlling reactor pressure at the desired value; and
- Observe reactor peak power and pressure

15.3.5.2.3 Systems Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems unless stated otherwise.

All plant control systems maintain normal operation unless specifically designated to the contrary, except that failure of all turbine bypass valves is assumed for the entire event.

15.3.5.3 Core and System Performance

15.3.5.3.1 Input Parameters and Initial Conditions

The turbine electrohydraulic control system (EHC) detects load rejection before a measurable turbine speed change takes place.

The closure characteristics of the TCVs are assumed conservatively such that the valves operate in the full arc (FA) mode. The TCVs have a full stroke closure time, from fully open to fully closed, from 0.15 seconds to 0.20 seconds. The worst case value (see Table 15.3-6) is assumed in the analysis.

The pressurization and/or the reactor scram may compress the water level to the low level trip setpoint (Level 2) and initiate the CRD high pressure makeup function, MSIV closure, and isolation condensers. Should this occur, it would follow sometime after the primary concerns of fuel thermal margin and overpressure effects have occurred.

15.3.5.3.2 Results

The results are shown in Figure 15.3-5.

This event will be analyzed for the plant-specific initial core configuration.

15.3.5.4 Barrier Performance

Peak absolute pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. The peak vessel bottom absolute pressure remains below its ASME code faulted pressure limit. In this event, the number of fuel rods that enter transition boiling is bounded by 1000 rods. It was assumed that all rods entering transition boiling fail.

15.3.5.5 Radiological Consequences

The off site dose for this event is less than 2.5 REM Total Effective Dose Equivalent (TEDE) assuming the bounding number (1000 rods) of fuel failures.

15.3.6 Turbine Trip With Total Turbine Bypass Failure

15.3.6.1 Identification of Causes

A variety of turbine or nuclear system malfunctions initiate turbine trips. Some examples are high velocity separator drain tank high levels, large vibrations, operator lockout, loss of control fluid pressure, low condenser vacuum and reactor high water level.

After the main turbine is tripped, turbine bypass valves are opened in their fast opening mode by the SB&PC system. As presented in Subsection 15.2.4.2, any single failures can only cause one bypass valve to fail to open on demand. Only multiple failures can cause all bypass valves fail to open on demand.

15.3.6.2 Sequence of Events and System Operation

15.3.6.2.1 Sequence of Events

Turbine trip at high power with failure of all bypass valves produces the sequence of events listed in Table 15.3-7.

15.3.6.2.2 Identification of Operator Actions

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure
- Verify that the generator breaker is automatically opened to allow electrical buses originally supplied by the generator to be supplied by the incoming power;
- Monitor reactor water level and pressure;
- Check turbine for proper operation of all auxiliaries during coastdown;
- Manually initiate ICS, if necessary, to control reactor pressure;
- Depending on conditions, maintain pressure for restart purposes, or initiate normal operating procedures for cooldown;
- Put the mode switch in the startup position before the reactor pressure decays to below 6 MPa;
- Cool down the reactor per standard procedure if a restart is not intended; and
- Investigate the cause of the trip, make repairs as necessary, and complete the scram report

15.3.6.2.3 Systems Operation

All plant control systems maintain normal operation unless specifically designated to the contrary, except that failure of all main turbine bypass valves is assumed for the entire transient time period analyzed. Credit is taken for successful operation of the RPS.

15.3.6.3 Core and System Performance

15.3.6.3.1 Input Parameters and Initial Conditions

Turbine stop valves full stroke closure time is in the range of 0.10 second to 0.15 second. The worst case (see Table 15.3-7) is assumed in the analysis. A reactor scram is initiated by the turbine stop valve position switch, and after the confirmation of no availability of the turbine bypass.

15.3.6.3.2 Results

A turbine trip with failure of the bypass system is simulated at 100% NBR power conditions in Figure 15.3-6.

The severity of this transient is similar to the generator load rejection with failure of bypass event presented in Subsection 15.3.5. This event does not have to be reanalyzed for a specific core configuration.

15.3.6.4 Barrier Performance

Peak absolute pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. The peak absolute pressure at the vessel bottom remains below its ASME Code faulted pressure limit. In this event, the number of fuel rods which enter transition boiling is bounded by 1000 rods. It was assumed that all rods entering transition boiling fail.

15.3.6.5 Radiological Consequences

The off site dose for this event is less than 2.5 REM Total Effective Dose Equivalent (TEDE) assuming the bounding number (1000 rods) of fuel failures.

15.3.7 Control Rod Withdrawal Error During Refueling

15.3.7.1 Identification of Causes

The event considered here is inadvertent criticality due to the complete withdrawal or removal of the most reactive rod during refueling. The probability of the initial causes, alone, is considered low enough to warrant its being categorized as an infrequent incident, because there is no postulated set of circumstances that results in an inadvertent control rod withdrawal error while in the REFUEL mode.

15.3.7.2 Sequence of Events and Systems Operation

Initial Control Rod Removal or Withdrawal

During refueling operation, system interlocks provide assurance that inadvertent criticality does not occur because a control rod has been removed or is withdrawn.

Fuel Insertion with Control Rod Withdrawn

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods be fully inserted when fuel is being loaded into the core. This requirement is backed up by refueling interlocks on rod withdrawal and movement of the refueling platform.

When the mode switch is in the REFUEL position, the interlocks prevent the platform from being moved over the core if a control rod is withdrawn and fuel is on the hoist. Likewise, if the refueling platform is over the core and fuel is on the hoist, control rod motion is blocked by the interlocks.

Second Control Rod Removal or Withdrawal

When the platform is not over the core (or fuel is not on the hoist) and the mode switch is in the REFUEL position, only one control rod can be withdrawn when the RC&IS SINGLE/GANG rod selection status is in the SINGLE rod selection mode. When the RC&IS SINGLE/GANG rod selection status is in the GANG rod selection mode, only one control rod pair with the same HCU may be withdrawn. Any attempt to withdraw an additional rod results in a rod block by the RC&IS interlock. Because the core is designed to meet shutdown requirements with one control rod pair (with the same HCU) or one rod of maximum worth withdrawn, the core remains subcritical even with one rod or rod pair withdrawn.

Control Rod Removal Without Fuel Removal

The design of the control rod, incorporating the bayonet coupling system does not physically permit the upward removal of the control rod without decoupling by rotation and the simultaneous or prior removal of the four adjacent fuel bundles.

Identification of Operator Actions

No operator actions are required to preclude this event, because the protection system design, as previously presented, prevents its occurrence.

15.3.7.3 Core and System Performance

Because the possibility of inadvertent criticality during refueling is precluded, the core and system performances are not analyzed. The withdrawal of the highest worth control rod during refueling does not result in criticality. This is verified experimentally by performing shutdown margin checks (see Section 4.3 for a description of the methods and results of the shutdown margin analysis). Additional reactivity insertion is precluded by refueling interlocks. Because no fuel damage can occur, no radioactive material is released from the fuel. Therefore, this event is not reanalyzed for specific core configurations.

15.3.7.4 Barrier Performance

An evaluation of the barrier performance is not made for this event because there is no postulated set of circumstances for which this event could occur.

15.3.7.5 Radiological Consequences

An evaluation of the radiological consequences is not made for this event, because no radioactive material is released from the fuel.

15.3.8 Control Rod Withdrawal Error During Startup

15.3.8.1 Identification of Causes

It is postulated that during a reactor startup, a gang of control rods or a single control rod is inadvertently withdrawn continuously due to a procedural error by the operator or a malfunction of the automated rod movement control system

The Rod Control and Information System (RC&IS) has a dual channel rod worth minimizer function that prevents withdrawal of any out-of-sequence rods from 100% control rod density to 50% control rod density (i.e., for Group 1 to Group 4 rods). It also has ganged withdrawal sequence restrictions at less than 50% control rod density such that, if the specified withdraw sequence constraints are violated, the rod worth minimizer function of the RCIS will initiate a rod block. These rod worth minimizer rod pattern constraints are in effect from 50% control rod density to the low power setpoint

The startup range neutron monitor (SRNM) has a period-based trip function that stops continuous rod withdrawal by initiating a rod block if the flux excursion, caused by rod withdrawal, generates a period shorter than 20 seconds. The period-based trip function also initiates a scram if the flux excursion generates a period shorter than 10 seconds. Any single SRNM rod block trip initiates a rod block. Any two divisional scram trips out of four divisions initiates a scram. The SRNM also has upscale rod block and upscale scram functions as a double protection for flux excursion. A detailed description of the period-based trip function is presented in Chapter 7

For this transient to happen, a large reactivity addition must be introduced. The reactor must be critical, with control rod density greater than 50%. Additionally, rod block logic of both rod worth minimizer channels must fail such that a gang of rods (or a single rod) can be continuously withdrawn. The causes of the event are summarized in Figure 15.3-7a. The probability for this event to occur is considered low enough to warrant its being categorized as a infrequent event.

15.3.8.2 Sequence of Events and Systems Operation

15.3.8.2.1 Sequence of Events

The sequence of events for a continuous control rod withdrawal error during reactor startup is shown in Table 15.3-9.

15.3.8.2.2 Identification of Operator Actions

No operator actions are required to terminate this event, because the SRNM period-based trip functions will initiate and terminate this event

15.3.8.3 Core and System Performance

15.3.8.3.1 Analysis Method and Analysis Assumptions

A BWR core response to RWE during startup was analyzed using the reactivity insertion analysis model described in Reference 15.3-4. It is a two-dimensional adiabatic model assuming no heat transfer to the coolant. The analysis consists of three steps. In Step 1, with the error rods being continuously withdrawn from full-in, the model is used to calculate the average power and period change as a function of time with a continuous reactivity insertion simulating the RWE event. In Step 2, the power versus time data are used as input to a calculation of the SRNM rod block and scram trip times. Both the rod block trip and scram trip times are then determined. In Step 3, the reactivity insertion input to the adiabatic model is adjusted such that after the period reaches the rod block setpoint (20 s), there is no further reactivity insertion. The RWE transient is then recalculated by the model with the adjusted reactivity input. The reactor scram time is also adjusted based on the time determined in Step 2. The calculated fuel enthalpy does not consider local peaking effect. In Step 4, the peak fuel enthalpy that includes the local peaking effect is calculated. Other assumptions used in the analysis are:

- The standard BWR data of the adiabatic model is used
- The scram reactivity shape is derived from the design core, assuming no failing rods and same scram speed for all rods
- Six delayed neutron groups are assumed
- An ABWR core with 3.5 m active core length fuel length was used. Application of this result to ESBWR is justified by the large margins shown below, the results will be confirmed for ESBWR in the COL submittal, using the initial core design and rod grouping.

15.3.8.3.2 Analysis Conditions and Results

Analysis Conditions

- The reactor is assumed to be in the critical condition before the control rod withdrawal, with an initial power of 0.001% rated, and a temperature of 286°C at the fuel cladding surface
- The worth of the withdrawn rods (gang) is 3% delta k from full-in to full-out. Gang rod withdrawal is used during a normal startup and provides a larger reactivity change than a single rod withdrawal case
- The control rod withdrawal speed is 30 mm/s, the nominal FMCRD withdrawal speed
- With the gang rod withdrawal, the reactor period monitored by any SRNM is relatively the same. Any single channel bypass of the SRNM does not affect the result

Analysis Result

With the reactivity insertion, based upon a gang worth of 3% delta k, the flux excursion generates a period of approximately 4 seconds. The rod block trip is initiated at 14 seconds after the start of the transient. The scram is initiated at about 25 seconds. The event is terminated by the scram. The peak fuel enthalpy reached is approximately 69.5 J/g, which is 0.63 J/g higher than the initial fuel enthalpy. The result is illustrated in Figure 15.3-7b

15.3.8.3.3 Evaluation Based On Criteria

Due to the effective protection function of the period-based trip function, the fuel enthalpy increase is small. The criterion of 711 J/g (170 cal/gm) for fuel enthalpy increase under RWE event is satisfied. An additional analysis was performed with the same assumptions and conditions as stated above, but assuming no protection function from the SRNM. Under this condition, the APRM setdown scram trip at 15% power provides the protection function. Flux and power excursion caused by continuous rod withdrawal error reaches the 15% power scram level and the reactor scrams. The result showed that the final peak fuel enthalpy was approximately 146.5 J/g, much lower than the RWE criteria of 711 J/g.

15.3.8.4 Barrier Performance

An evaluation of the barrier performance is not made for this event, because there is no fuel damage in this event and only with mild change in gross core characteristics

15.3.8.4.1 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, because no radioactive material is released from the fuel

15.3.8.4.2 COL Action Item:

Perform confirmatory analysis with the ESBWR initial core and rod group assignments.

15.3.9 Control Rod Withdrawal Error During Power Operation

15.3.9.1 Identification of Causes

In ESBWR, the automated thermal limit monitor (ATLM) subsystem performs the associated rod block monitoring function. The ATLM is a dual channel subsystem of the RC&IS. Each ATLM channel has two independent thermal limit monitoring functions. One function monitors the minimum critical power ratio (MCPR) limit and protects the operating limit MCPR, another function monitors the maximum linear heat generation rate (MLHGR) limit and protects the operating limit of the MLHGR. The rod block algorithm and setpoint of the ATLM are based on actual on-line core thermal limit information. If any operating limit protection setpoint limit is reached, such as due to control rod withdrawal, control rod withdrawal permissive is removed. Detailed description of the ATLM subsystem is presented in Chapter 7.

The causes of a potential RWE are either a procedural error by the operator in which a single control rod or a gang of control rods is withdrawn continuously, or a malfunction of the automated rod withdrawal sequence control logic during automated operation in which a gang of control rods is withdrawn continuously. But in either case, the operating thermal limits rod block function blocks any further rod withdrawal when the operating thermal limit is reached. That is, the withdrawal of rods is stopped before the operating thermal limit is reached. Because there is no operating limit violation due to the preventive function of the ATLM, there is no rod withdrawal error transient event.

15.3.9.2 Sequence of Events and System Operation

Due to an operator error or a malfunction of the automated rod withdrawal sequence control logic, a single control rod or a gang of control rods is withdrawn continuously. The ATLM operating thermal limit protection function of either the MCPR or MLHGR protection algorithms stops further control rod withdrawal when either operating limit is reached. There is no basis for occurrence of the continuous control rod withdrawal error event in the power range.

No operator action is required to preclude this event, because the plant design as described above prevents its occurrence.

15.3.9.3 Core and System Performance

The performance of the ATLM subsystem of the RC&IS prevents the RWE event from occurring. The core and system performance are not affected by such an operator error or control logic malfunction. There is no need to analyze this event.

15.3.9.4 Barrier Performance

An evaluation of the barrier performance is not made for this event, because there is no postulated set of circumstances for which this event could occur.

15.3.9.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, because no radioactive material is released from the fuel.

15.3.10 Fuel Assembly Loading Error, Mislocated Bundle

The mislocated fuel bundle error involves the mislocation of at least two fuel bundles. One location is loaded with a bundle that would potentially operate at a lower power than it would otherwise. The other location would operate at a higher power.

There is a strong possibility that the core monitor will recognize the mislocated fuel bundle, thereby allowing the reactor operators to mitigate the consequences of this event. In the best situation where the high radial power mislocated bundle is adjacent to an instrument, the power adjustment in radially automatic fixed in-core probe (AFIP) or local power range monitor (LPRM) adapting monitoring systems will detect the higher bundle power. The reactor will be operated such that the most limiting of the bundles near the mislocation will be maintained below the operating limit MCPR. A less effective situation is where the mislocated bundle has a bundle between it and an instrument.

An ineffective situation occurs when the core monitor does not recognize the mislocation because the monitoring system is not radially AFIP or LPRM adapted.

Assuming the mislocated bundle is not monitored, one possible state of operation for the fuel bundle is that it operates through the cycle close to or above the fuel thermal mechanical limit.

The potential exists that if the fuel bundle operates above the thermal mechanical limit, one or more fuel rods may experience cladding failure. If this were to occur, the adverse consequence of operation are detectable and can be suppressed during operation just like leaking fuel rods resulting from other causes. In this context, the adverse consequence is the perforation of a small number of fuel rods in the mislocated fuel assembly. Any perforations that may result would be localized, there would be only a few perforations, and the perforations would not propagate to other fuel rods or fuel assemblies. The perforation of a small number of fuel rods leads to the release of fission products to the reactor coolant, which are detected by the offgas system. A control rod inserted in the vicinity of the leaking fuel rods suppresses the power in the leaking fuel rods, returns the thermal–hydraulic condition to normal heat transfer with its characteristic low temperature difference between the cladding and the coolant, and reduces the fission product release and offgas.

Further discussion on the analysis methods for the mislocated bundle event is given in Reference 15.3-3

Proper location of the fuel assembly in the reactor core is readily verified by visual observation and assured by verification procedures during core loading. GE provides recommended fuel assembly loading instructions for the initial core as part of the Startup Test Instructions (STIs). It is expected that the plant owners use similar procedures during subsequent refueling operations. Verification procedures include inventory checks, current bundle location logs, serial number verifications and visual or photographic inspection of the loaded core. The verification procedures are designed to minimize the possibility of the occurrence of the mislocated bundle event.

15.3.11 Fuel Assembly Loading Error, Misoriented Bundle

The misoriented bundle event has been evaluated in Reference 15.3-3, on a generic bounding basis.

Proper orientation of fuel assemblies in the reactor core is readily verified by visual observation and assured by verification procedures during core loading. Five separate visual indications of proper fuel assembly orientation exist:

- The channel fastener assemblies, including the spring and guard used to maintain clearances between channels, are located at one corner of each fuel assembly adjacent to the center of the control rod.
- The identification boss on the fuel assembly handle points toward the adjacent control rod.
- The channel spacing buttons are adjacent to the control rod passage area.
- The assembly identification numbers that are located on the fuel assembly handles are all readable from the direction of the center of the cell.
- There is cell–to–cell replication.

Experience has demonstrated that these design features are clearly visible so that any misoriented fuel assembly would be readily identifiable during core loading verification.

The bounding analysis for the misoriented fuel assembly is discussed in detail in Reference 15.3-1.

Bounding radiological analysis of these events is contained in Reference 15.3-3. Individual plants must verify periodically (once every few years) that they are within the limiting site meteorological criteria contained in Reference 15.3-3 to ascertain that they are within the limits, or they will be required to demonstrate no fuel failure occurs for these two events, i.e., analyze these events as AOOs (see Reference 15.3-1).

15.3.12 Inadvertent SDC Function Operation

15.3.12.1 Identification of Causes

A shutdown cooling malfunction leading to a moderate temperature decrease could result from misoperation of the cooling water controls for the RWCU/SDC system heat exchangers. The resulting temperature decrease would cause a slow insertion of positive reactivity into the core. If the operator did not act to control the power level, the system would settle at a new steady state without violating fuel thermal limits and without any measurable increase in nuclear system pressure.

15.3.12.2 Sequence of Events and Systems Operation

Sequence of Events

A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the cooling water controls for RWCU/SDC system heat exchangers. The resulting temperature decrease causes a slow insertion of positive reactivity into the core. During power operation, the system settles at a new steady state. During startup, scram may occur. In either case, no thermal limits are reached. The sequence of events for this event is a slow rise in reactor power. The operator may take action to limit the power rise. Flux scram occurs if no operator action is taken.

System Operation

No unique safety actions are required to avoid unacceptable safety results for transients as a result of a reactor coolant temperature decrease induced by misoperation of the shutdown cooling heat exchangers. In startup or cooldown operation, where the reactor is at or near critical, the slow power increase resulting from the cooler moderator temperature is controlled by the operator in the same manner normally used to control power in the startup range.

15.3.12.3 Core and System Performance

The increased subcooling caused by misoperation of the RWCU/SDC shutdown cooling mode could result in a slow power increase due to the reactivity insertion. At power the reactor settles in a new steady state. During startup or shutdown, this power rise is terminated by a flux scram before fuel thermal limits are approached. Therefore, only a qualitative description is provided here and this event does not have to be analyzed for a specific core configuration.

15.3.12.4 Barrier Performance

As previously presented, the effects of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.3.12.5 Radiological Consequences

Because this event does not result in any fuel failures, no analysis of radiological consequences is required for this event

15.3.13 Inadvertent Opening of a Safety-Relief Valve

15.3.13.1 Identification of Causes

Cause of inadvertent safety-relief valve (SRV) opening is attributed to malfunction of the valve or an operator initiated opening. It is therefore simply postulated that a failure occurs and the event is analyzed accordingly. Detailed discussion of the valve design is provided in Chapter 5.

During normal operation a spurious signal causes one SRV to open. The steam of this SRV is discharged in the suppression pool, if the subsequent manual closure of the SRV is not obtained, then the suppression pool will increase its temperature, reaching the scram set-point and finally scramming the reactor.

15.3.13.2 Sequence of Events and Systems Operation

Sequence of Events

Table 15.3-11 lists the sequence of events for this event.

Identification of Operator Actions

The plant operator must re-close the valve as soon as possible and check that reactor and TG output return to normal. If the valve cannot be closed, plant shutdown should be initiated.

Systems Operation

This event assumes normal functioning of normal plant instrumentation and controls, specifically the operation of the pressure regulator and level control systems.

15.3.13.3 Core and System Performance

The opening of one SRV allows steam to be discharged into the suppression pool. The sudden increase in the rate of steam flow leaving the reactor vessel causes a mild depressurization transient.

The SB&PC system senses the nuclear system pressure decrease and within a few seconds closes the TCVs far enough to stabilize the reactor vessel pressure at a slightly lower value and the reactor settles at nearly the initial power level. Eventually, the plant automatically scrams on high suppression pool temperature.

Thermal margins decrease only slightly through the transient and no fuel damage results from the event.

15.3.13.4 Barrier Performance

The transient resulting from the inadvertent SRV open is a mild depressurization which is within the range of normal load following and therefore has no significant effect on RCPB and containment design pressure limits.

15.3.13.5 Radiological Consequences

While the effect of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool. Because this activity is contained in the primary containment, there is no exposure to operating personnel. Because this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment, use the fuel and auxiliary pools cooling system (FAPCS) to remove radioactivity from the pool, or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release shall be in accordance with the established Technical Specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.3.14 Inadvertent Opening of a Depressurization Valve

15.3.14.1 Identification of Causes

Potential causes of inadvertent Depressurization Valve (DPV) opening are malfunction of the valve or an operator initiated opening. It is therefore simply postulated that a failure occurs and the event is analyzed accordingly. Detailed discussion of the valve design is provided in Chapter 5.

15.3.14.2 Systems Operation and Sequence of Events

Identification of Operator Actions

The operator should monitor the reactor water level and control makeup systems. Because the valve cannot be closed remotely, plant shutdown should be initiated. The operator should confirm the plant scrams on drywell pressure, or scram before the high drywell pressure is reached if a DPV is confirmed to be open, and monitor RPV and DW pressure and control with the bypass valves if necessary.

Systems Operation

This event assumes normal functioning of normal plant instrumentation and controls, specifically the operation of the pressure regulator and level control systems.

15.3.14.3 Core and System Performance

The opening of one DPV allows steam to be discharged into the drywell. The sudden increase in the rate of steam flow leaving the reactor vessel causes a mild depressurization transient.

The SB&PC system senses the nuclear system pressure decrease and within a few seconds closes the TCVs far enough to stabilize the reactor vessel pressure at a slightly lower value and the reactor settles at nearly the initial power level. Eventually, the plant automatically scrams on high drywell pressure. After scram, depressurization of the RPV will resume.

Thermal margins decrease only slightly through the transient and no fuel damage results from the event.

15.3.14.4 Barrier Performance

The transient resulting from the inadvertent DPV open is a mild depressurization which is within the range of normal load following and therefore has no significant effect on RCPB and containment design pressure limits.

15.3.14.5 Radiological Consequences

While the effect of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the drywell. After the drywell pressurizes, and the DW to WW vents clear, the steam will be vented to suppression pool and condense in the pool. Because the activity is contained in the primary containment, there is no exposure to operating personnel. Because this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment, use FAPCS to remove radioactivity from the pool, or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release shall be in accordance with the established Technical Specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.3.15 Stuck Open Safety-Relief Valve

15.3.15.1 Identification of Causes

Cause of a stuck open safety-relief valve is attributed to the malfunction of the valve after it has opened either inadvertently or in response to a high pressure signal. It is, therefore, simply postulated that a failure occurs and the event is analyzed accordingly. Detailed discussion of the valve design is provided in Chapter 5.

In this analysis, after any event which produces the scramming of the reactor, it is assumed that a SRV remains open without any possibility of closure. The operations of the ICs produce a depressurization, with the HP_CRD operating to recover the level after the Scram. Finally the reactor reaches pressure near atmospheric.

15.3.15.2 Sequence of Events and Systems Operation

Sequence of Events

Table 15.3-12 lists the sequence of events for this event.

Identification of Operator Actions

The plant operator must re-close the valve as soon as possible and check that the reactor and TG output return to normal. If the valve cannot be closed and the reactor has scrammed because of some other reason (if the SRVs are in the open condition the plant must necessarily have scrammed previously, except for the Inadvertent SRV opening analyzed previously), manual activation of the IC and other systems must be initialized to reduce the amount of steam reaching the suppression pool.

Systems Operation

This event assumes normal functioning of the plant instrumentation and controls, specifically the operation of the pressure regulator and water level control systems.

15.3.15.3 Core and System Performance

The opening of one SRV allows steam to be discharged to the suppression pool. The sudden increase in the rate of steam flow leaving the reactor vessel causes a depressurization transient, with the vessel pressure slowly decreasing until reaching atmospheric pressure. The SRV steam discharge also results in a slight heating of the suppression pool.

Thermal margins decrease only slightly through the transient and no fuel damage is predicted for this event.

15.3.15.4 Barrier Performance

As presented previously, the transient resulting from a stuck open relief valve is the total depressurization of the pressure vessel, which is within the range of normal plant operation and therefore has no significant effect on RCPB and containment design pressure limits.

15.3.15.5 Radiological Consequences

While the effect of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool. Because this activity is contained in the primary containment, there is no exposure to operating personnel. Because this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity "bottled up" in the containment, use FAPCS to remove radioactivity from the pool, or discharge it to the environment in a controlled manner. If purging of the containment is chosen, the release shall be done in accordance with the plant's Technical Specifications. Consequently, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.3.16 Liquid Containing Tank Failure

15.3.16.1 Identification of Causes

An unspecified event causes the complete release of the radioactive inventory in all tanks containing radionuclides in the liquid radwaste system. Postulated events that could cause a release of the inventory of a tank are sudden unmonitored cracks in the vessel or operator error. Small cracks and consequent low level releases are bounded by this analysis and should be contained without any significant release.

The ESBWR Radwaste Building is designed to withstand all credible seismic events. In addition, the walls of all compartments containing high level liquid radwaste are lined up to a height capable of containing the release of all the liquid radwaste in the compartment. Because of these design capabilities, it is considered remote that any major event involving the release of liquid radwaste into these volumes would result in the release of these liquids to the environment via the liquid pathway. Releases as a result of major cracks would instead result in the release of the liquid radwaste to the compartment and then to the building sump system for containment in other tanks or emergency tanks. A complete description of the liquid radwaste system is found in Section 11.2, except for the tank inventories, which are found in Section 12.2.

A liquid radwaste release caused by operator error is also considered a remote possibility. Operating techniques and administrative procedures emphasize detailed system and equipment operating instructions. A positive action interlock system is also provided to prevent inadvertent opening of a drain valve. Should a release of wastes occur, the lining would contain the release until the floor drain sump pumps in the building capture and contain such spills.

The probability of a complete tank release is considered low enough to warrant this event as an Infrequent Event.

15.3.16.2 Sequence of Events and Systems Operations

Following a failure, the area radiation alarms would be expected to alarm at one minute with operator intervention following at approximately five minutes after release. However, the rupture of a waste tank would leave little recourse for the operator. Gaseous wastes would be trapped following alarm initiation, because isolation would occur upon alarm initiation. However, no credit for isolation is taken for this aspect and gaseous releases are assumed to be purged to the environment.

Liquid release would be contained within the liner and would present no immediate threat to the environment leaving the operator sufficient time (on the order of hours) in which to recover systems to pump the release into holding tanks or emergency tanks.

15.3.16.3 Results

A single pathway is considered for release of fission products to the environment via airborne releases. The liquid pathway is not considered because of the mitigation capabilities of the Radwaste Building.

For the airborne pathway, volatile iodine species in the tank using the inventories in Section 12.2 are considered. These inventories are based upon the design basis release rates found in Section 12.2. Although isolation is expected within minutes of the occurrence, release of 10% of the iodine inventory is conservatively assumed over a two-hour period. Specific parameters for this analysis are found in Tables 15.3-12 and 15.3-13.

No liquid or significant (from airborne species) ground contamination is expected. Airborne doses are given in Table 15.3-14 and are a fraction of 10 CFR 100 criteria.

15.3.17 COL Information

Confirm the applicability of the Startup RWE analysis to the initial core design.

Confirm the applicability of the generic radiological dose assessment for misloaded fuel bundles to the site meteorological characteristics.

15.3.18 References

15.3-1 GE Nuclear Energy, "General Electric Standard Application for Reactor Fuel--United States Supplement," NEDE-24011-P-A-US, (Latest approved revision).

15.3-2 GE Nuclear Energy, "Radiological Accident Evaluation—The CONAC03 Code," NEDO-21143-1, December 1981.

15.3-3 FLN–2004–026, "GESTAR I Amendment 28 Revision 1, Misloaded Fuel Bundle Event Licensing Basis Change to Comply with Standard Review Plan 15.4.7," Margaret E. Harding to Mel B. Fields August 23, 2004

15.3-4 J. Paone and J. A. Woolley, "Rod Drop Accident Analysis for Large Boiling Water Reactors, Licensing Topical Report," March 1972 (NEDO-10527, Supplements 1 and 2).

Table 15.3-1

Results Summary of Infrequent Events

Sub-section I.D.	Description	Max. Neutron Flux, % NBR	Max. Dome Pressure, MPaG (psig)	Max. Vessel Bottom Pressure, MPaG (psig)	Max. Steamline Pressure, MPaG (psig)	Max. Core Average Surface Heat Flux, % of Initial	ΔCPR
15.3.1.1	Loss of Feedwater Heating with SCRRI failure	120.8	7.11 (1031)	7.25 (1051)	7.07 (1025)	120.8	0.15
15.3.2.1	FWCF – Maximum Demand	117.6	7.28 (1056)	7.42 (1076)	7.23 (1049)	108.6	0.07
15.3.2.2	Pressure Regulator Failure – Opening of all TCVs and BPVs	100	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	0.00
15.3.2.3	Pressure Regulator Failure – Closing of all TCVs and BPVs	137.9	8.10 (1175)	8.23 (1194)	8.10 (1175)	103.8	0.07
15.3.2.4	Load Rejection with total bypass failure	308.7	8.19 (1188)	8.32 (1207)	8.20 (1189)	107.8	0.16
15.2.2.5	Turbine Trip with total bypass failure	271.5	8.19 (1188)	8.32 (1207)	8.19 (1188)	106.6	0.15
15.2.2.6	Inadverted SRV open	100.5	7.08 (1027)	7.21 (1046)	7.04 (1021)	100.5	0.02
15.2.2.7	Stuck open SRV (1)	100.0	7.08 (1027)	7.21 (1046)	7.04 (1021)	100.0	0.0

(1) The initiating event can produce some over power, but the Stuck SRV open should not produce any appreciable overpower or MCPR reduction.

Table 15.3-2

Sequence of Events for Loss of Feedwater Heating With Failure of Selected Control Rod Run-In

Time (s)	Event
0	Initiate a 55.6°C (100°C) temperature reduction in the FW system.
25 (est.)	Initial effect of unheated FW starts to raise core power level.
80	High thermal simulated Scram is reached but it is not credited.
300 (est.)	New Steady State Reached.

Table 15.3-3

Sequence of Events for Feedwater Controller Failure – Maximum Demand

Time (sec)	Event *
0	Initiate simulated runout of all FW pumps (170% at rated vessel pressure).
12.3	Main turbine bypass valves opened to control vessel pressure.
15.3	L8 vessel level setpoint is reached
16.15	Scram, trip of main turbine and FW pump runback is activated.
16.17	Turbine Bypass activation limits the pressurization of the vessel.
16.35	The rods begin to enter inside the core
Later	L2 can be reached, activating IC and HP_CRD to recover the level

* See Figure 15.3-2

Table 15.3-4

Sequence of Events for Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

Time (sec)	**Events**
0	Simulate all turbine control valves and bypass valves to open.
17.89	Low turbine inlet pressure trip initiates main steamline isolation.
19.17	MSIV position switch at 85% initiates scram and activates the IC
22.89	Main steam isolation valves closed. Bypass valves remain open, exhausting steam in steamlines downstream of isolation valves.
28.14	L2 setpoint is reached
34.17	The IC begins to remove heat from the vessel
38.14	HP_CRD is activated, this recovers the level.

* See Figure 15.3-3

Table 15.3-5

Sequence of Events for Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

Time (sec)	**Event**
0	Simulate zero steam flow demand to main turbine and bypass valves.
0	Turbine control valves start to close.
1.78	Neutron flux reaches high flux scram setpoint and initiates a reactor scram.
2.03	The rods begin to enter inside the core
2.5	TCV is closed
Long term	HP_CRD is activated because L2 to recover the level

* See Figure 15.3-4

Table 15.3-6

Sequence of Events for Generator Load Rejection With Total Turbine Bypass Failure

Time (sec)	**Event**
(-)0.015	Turbine-generator detection of loss of electrical load.
0.0	Turbine-generator load rejection sensing devices trip to initiate turbine control valves fast closure.
0.0	Turbine bypass valves fail to operate.
0.08	Turbine control valves closed.
0.15	After detection of not enough bypass availability the RPS initiates a reactor scram.
0.40	The rods begin to enter inside the core
Long term	HP_CRD is activated on L2 to recover the level

* See Figure 15.3-5

Table 15.3-7

Sequence of Events for Turbine Trip With Total Turbine Bypass Failure

Time (sec)	**Event**
0.0	Turbine trip initiates closure of main stop valves.
0.0	Turbine bypass valves fail to operate.
0.10	Turbine stop valves close.
0.15	After detection of not enough bypass availability the RPS initiates a reactor scram.
0.40	The rods begin to enter inside the core
Long term	HP_CRD is activated on L2 to recover the level

* See Figure 15.3-6

Table 15.3-8

Sequence of Events for Continuous Control Rod Withdrawal Error During Reactor Startup

Time (sec)	Events
0	Operator withdraws a gang of rods (or a single rod) continuously; or a gang of rods (or single rod) is withdrawn continuously due to a malfunction of the Automated Rod Movement Control System
~6	Neutron flux increases rapidly due to the continuous reactivity addition, with a very short period
~14	The SRNM Period-Based Rod Block Trip initiates rod block due to short period (less than the 20-second setpoint)
~25	The SRNM Period-Based Scram Trip initiates reactor scram due to short period (less than the 10-second setpoint)
~27	Reactor is scrammed (all rods inserted) and the event is terminated

Table 15.3-9

Sequence of Events for the Mislocated Bundle

(1)	During the core loading operation, a bundle is loaded into the wrong core location.
(2)	Subsequently, the bundle designated for this location is incorrectly loaded into the location of the previous bundle.
(3)	During the core verification procedure, the two errors are not observed.
(4)	The plant is brought to full power operation without detecting misplaced bundles.
(5)	The plant continues to operate throughout the cycle.

Table 15.3-10

Sequence of Events for the Misoriented Bundle

(1)	During the core loading operation, a bundle is rotated and loaded with incorrect orientation.
(2)	During the core verification procedure, the orientation error is not observed.
(3)	The plant is brought to full power operation without detecting the misoriented bundle.
(4)	The plant continues to operate throughout the cycle.

Table 15.3-11

Sequence of Events for Inadvertent SRV Opening

Time (sec)	**Event**
0	Spurious opening of one SRV
1.0	Relief valve flow reaches full flow.
30.0	System establishes new steady-state operation.
412.5	Suppression pool temperature reaches the setpoint; suppression pool cooling function is initiated.
412.5	Suppression pool temperature reaches setpoint; reactor scram is automatically initiated.

See Figure 15.3-8

Table 15.3-12

Sequence of Events for Stuck Open Safety Relief Valve

Time (sec)	Event
0	Event happens, the reactor is scrammed and one SRV stuck open
10.0	Relief valve flow reaches one SRV flow.
10.0	The vessel begins depressurization
19.3	HP_CRD is activated
121.8	Low steamline pressure is activated
122.6	MSIV at 85%
124.8	MSIV is closed
154.0	ICs at full operation
Long term	Suppression pool temperature reaches the setpoint; suppression pool cooling function is initiated. Atmospheric pressure is reached

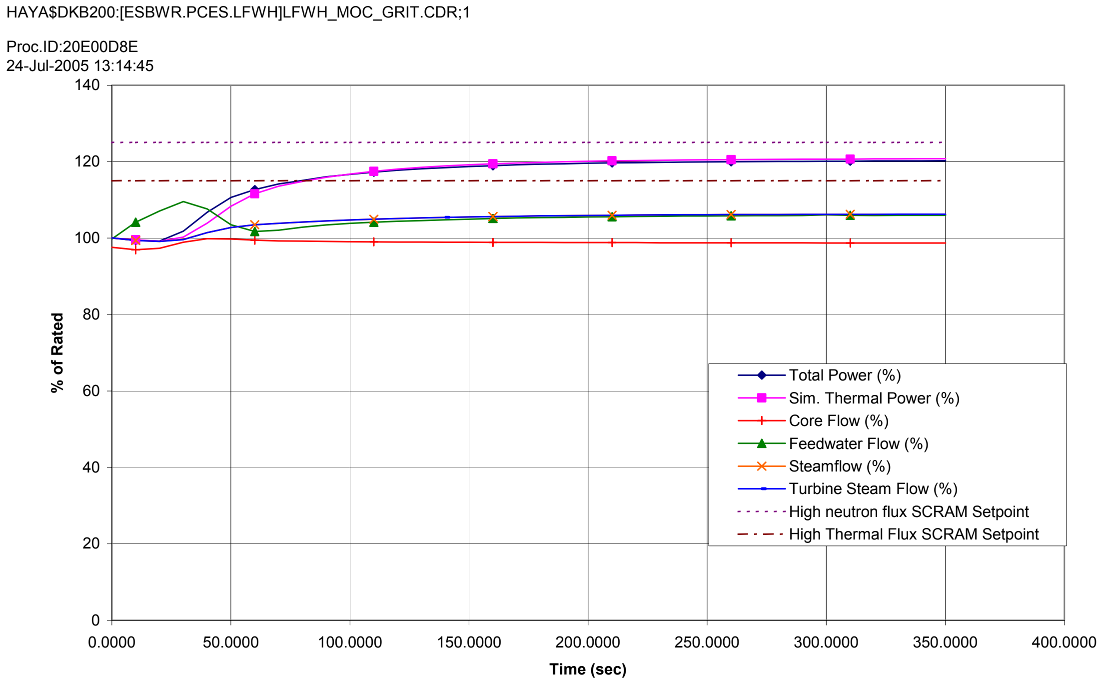


Table 15.3-13

Figure 15.3-1a. Loss of Feedwater Heating with SCRRI Failure

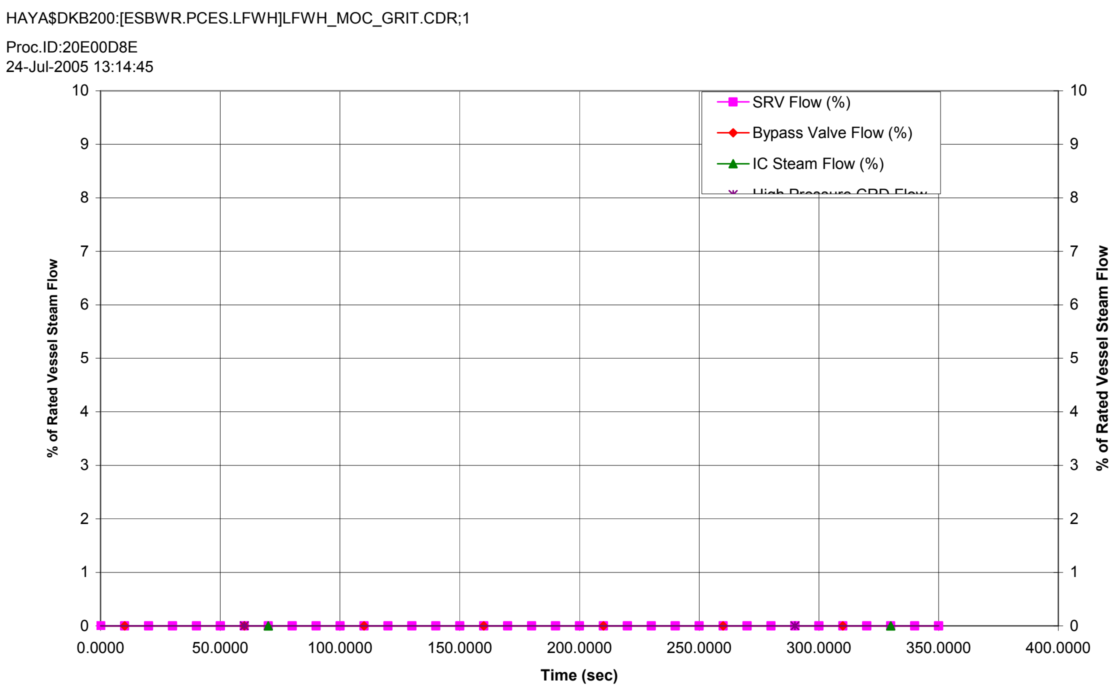


Figure 15.3-1b. Loss of Feedwater Heating with SCRRI Failure

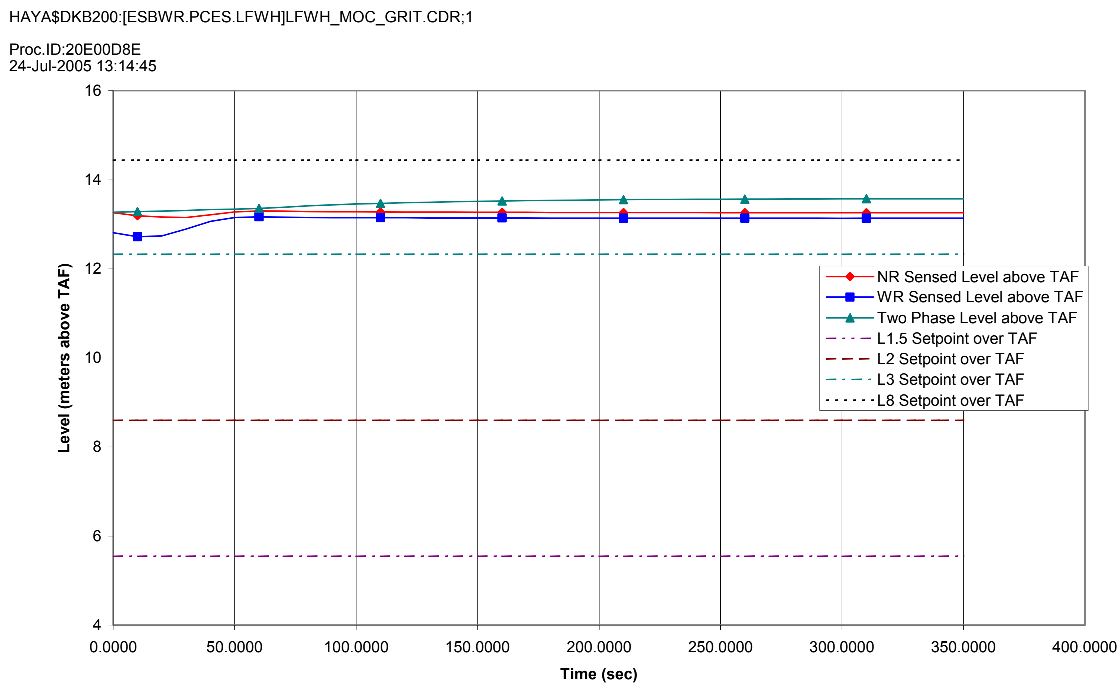


Figure 15.3-1c. Loss of Feedwater Heating with SCRRI Failure

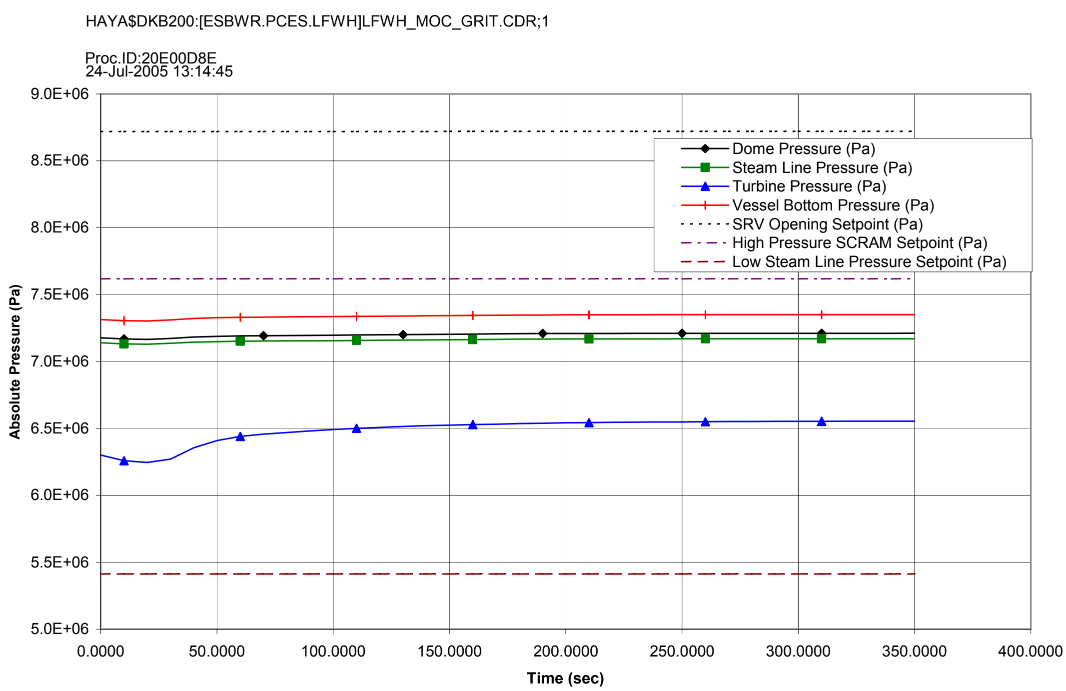


Figure 15.3-1d. Loss of Feedwater Heating with SCRRI Failure

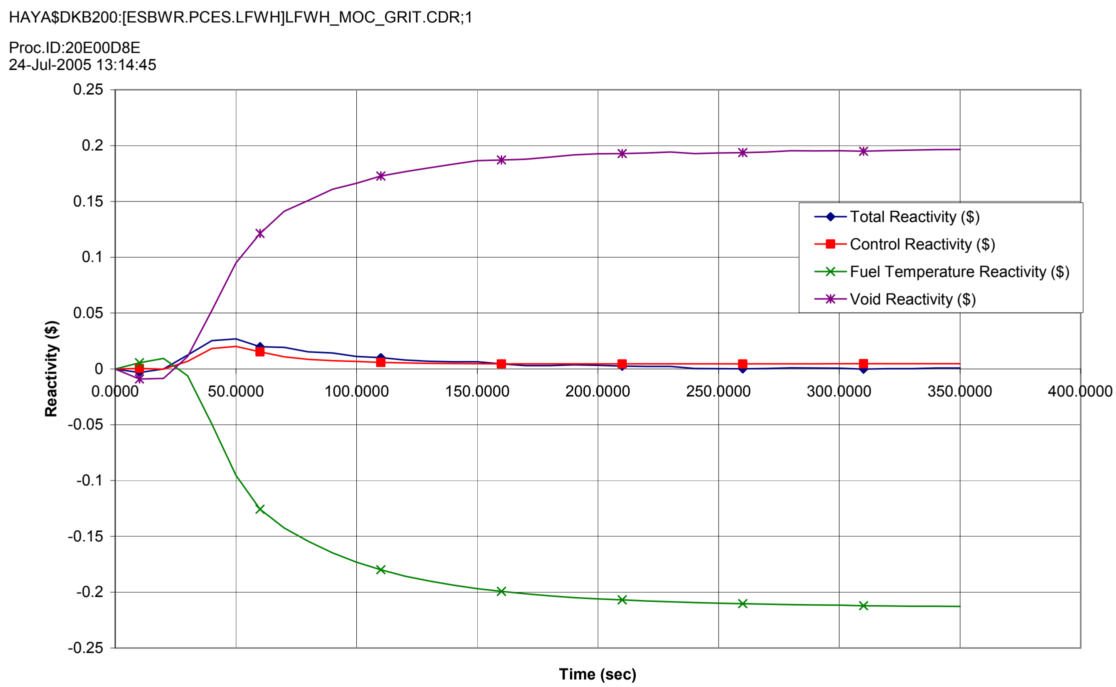


Figure 15.3-1e. Loss of Feedwater Heating with SCRRI Failure

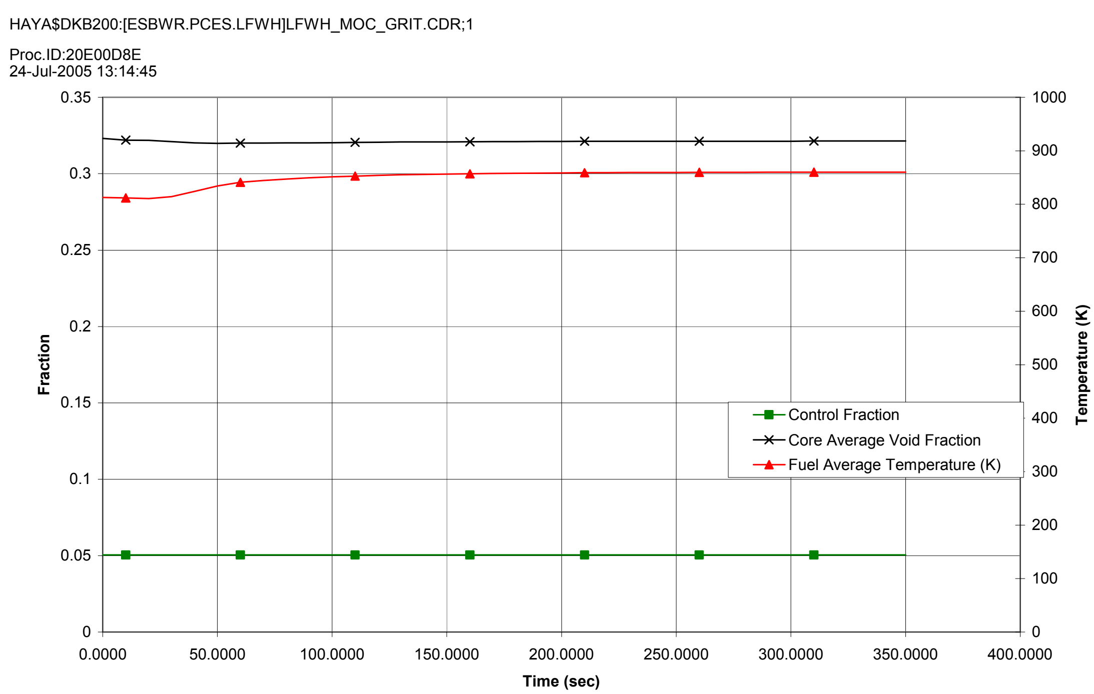


Figure 15.3-1f. Loss of Feedwater Heating with SCRRI Failure

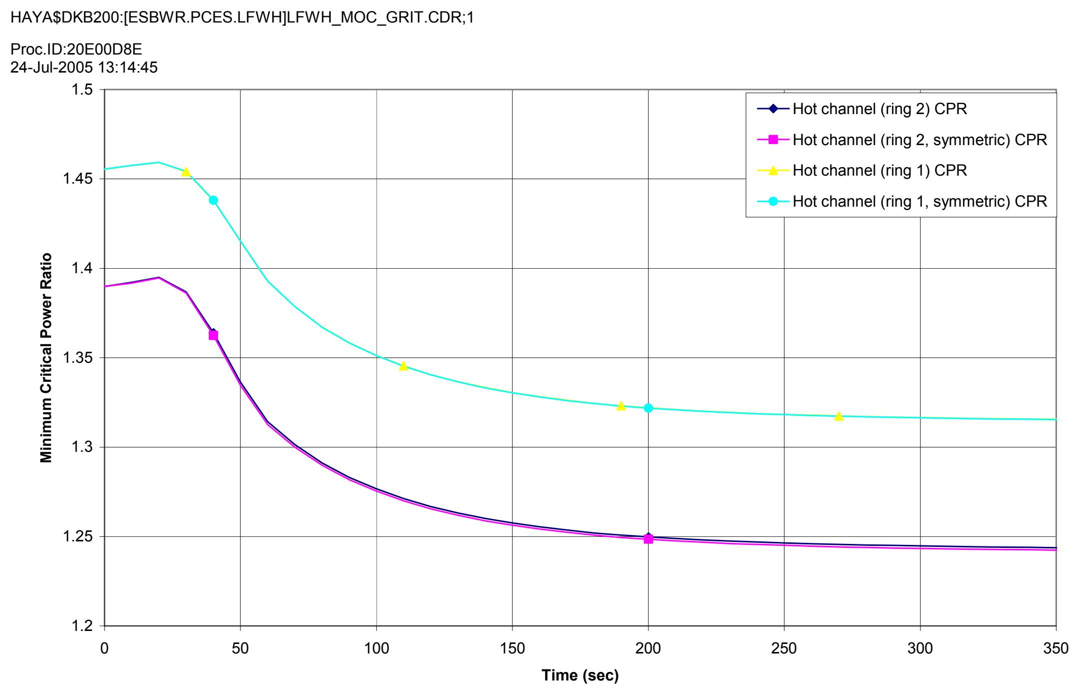


Figure 15.3-1g. Loss of Feedwater Heating with SCRRI Failure

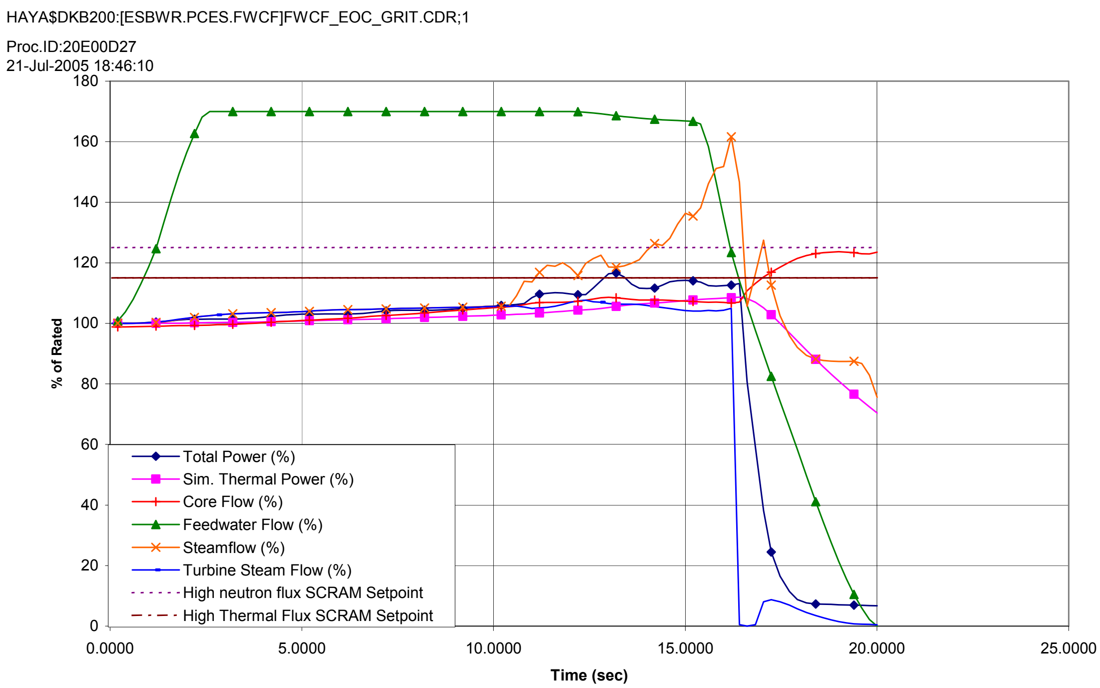


Figure 15.3-2a. Feedwater Controller Failure – Maximum Demand

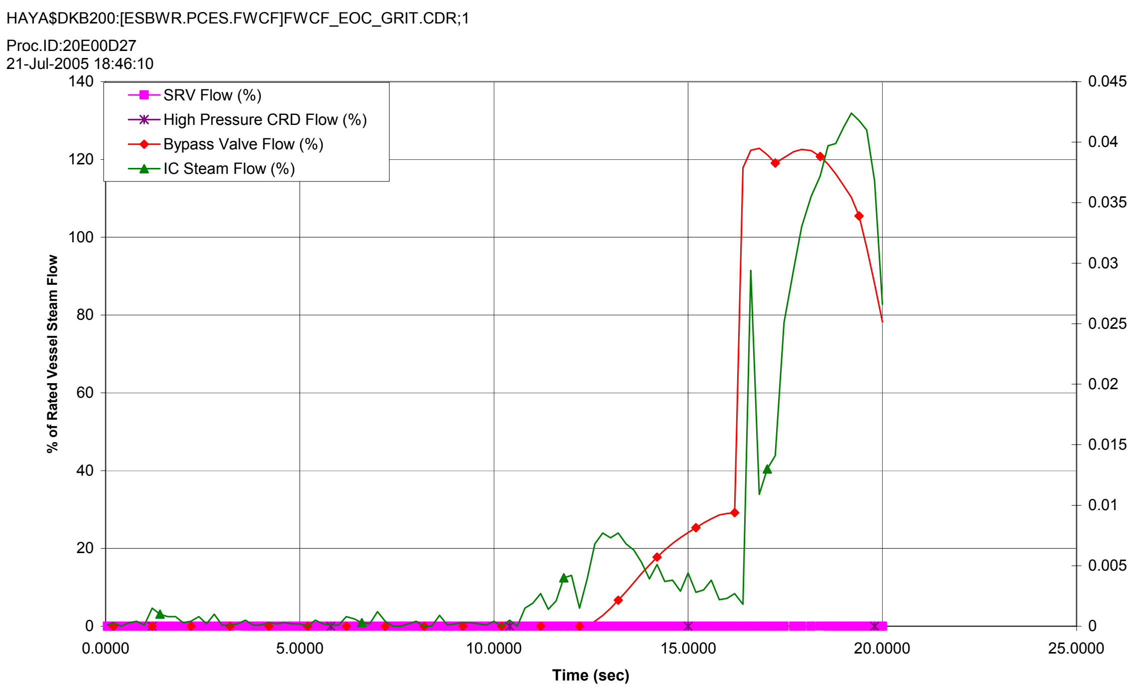


Figure 15.3-2b. Feedwater Controller Failure – Maximum Demand

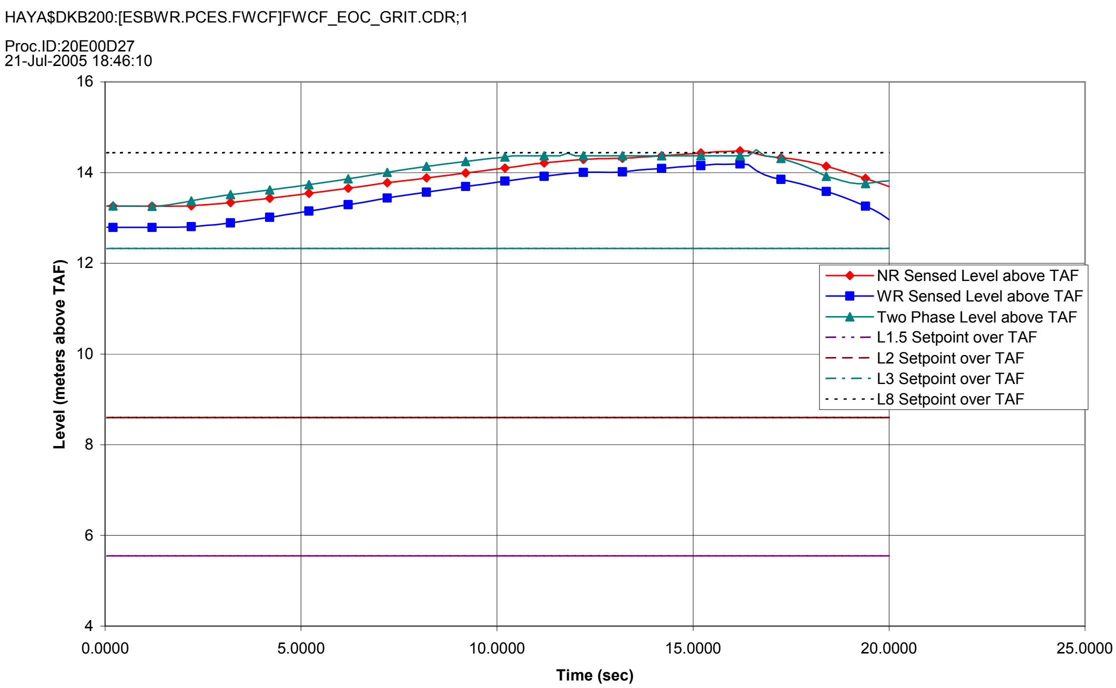


Figure 15.3-2c. Feedwater Controller Failure – Maximum Demand

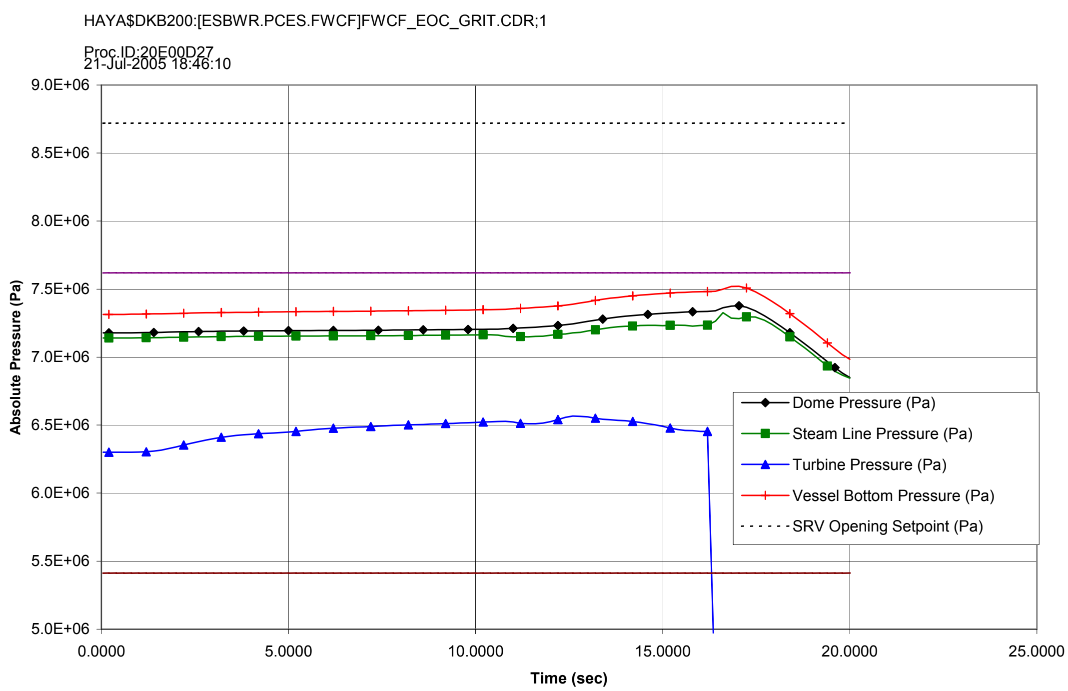


Figure 15.3-2d. Feedwater Controller Failure – Maximum Demand

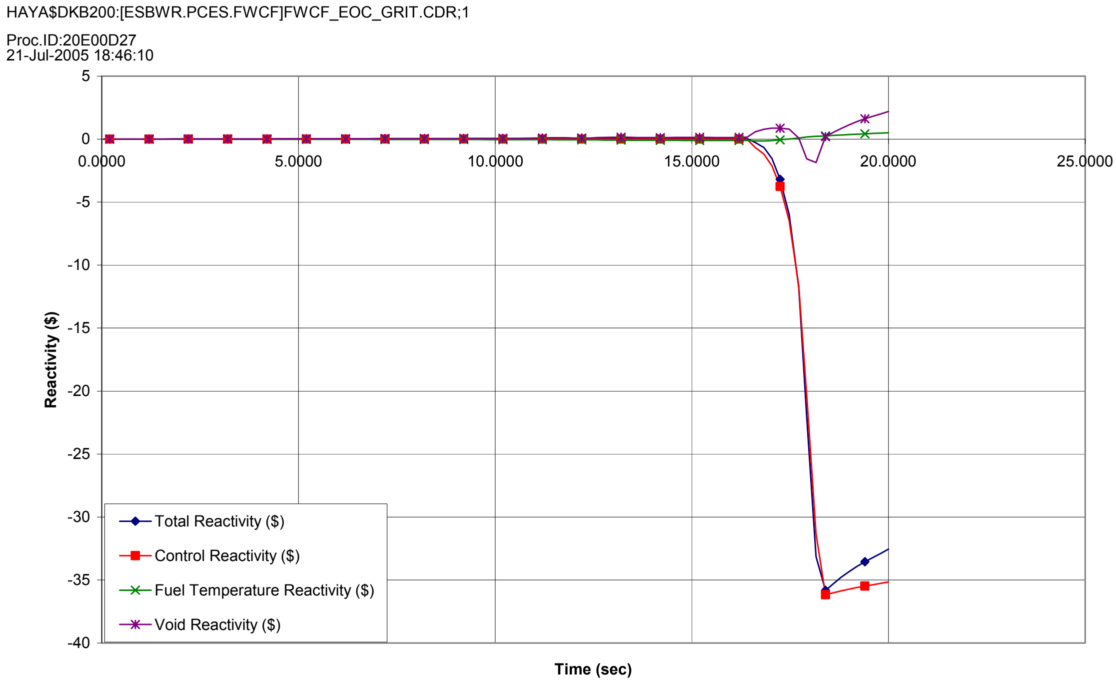


Figure 15.3-2e. Feedwater Controller Failure – Maximum Demand

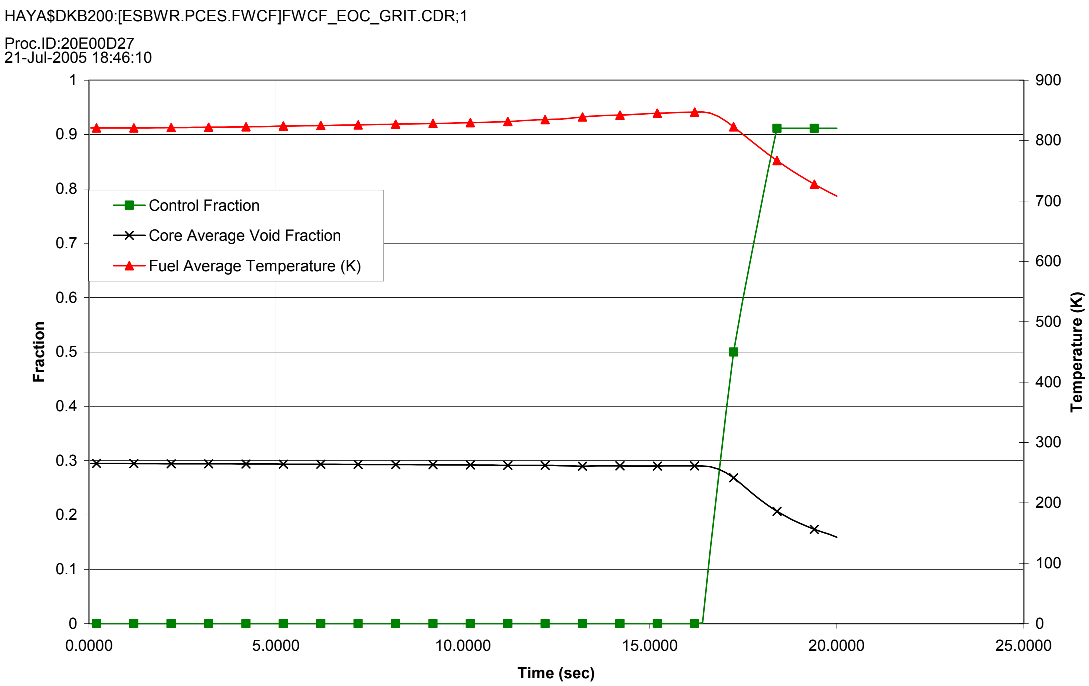


Figure 15.3-2f. Feedwater Controller Failure – Maximum Demand

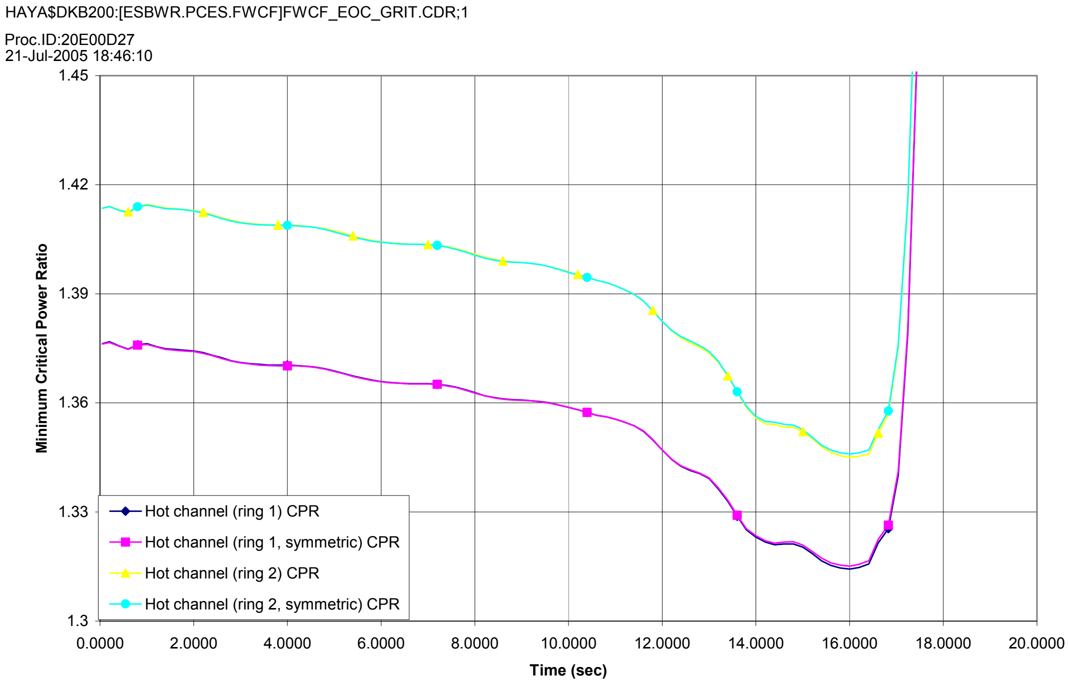


Figure 15.3-2g. Feedwater Controller Failure – Maximum Demand

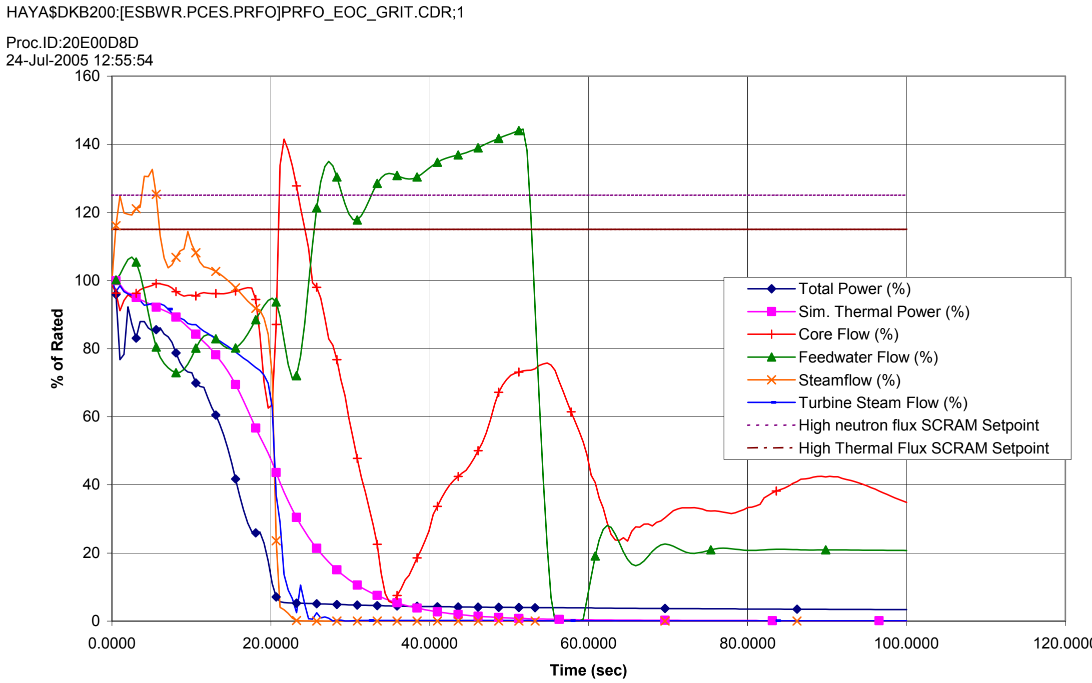


Figure 15.3-3a. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

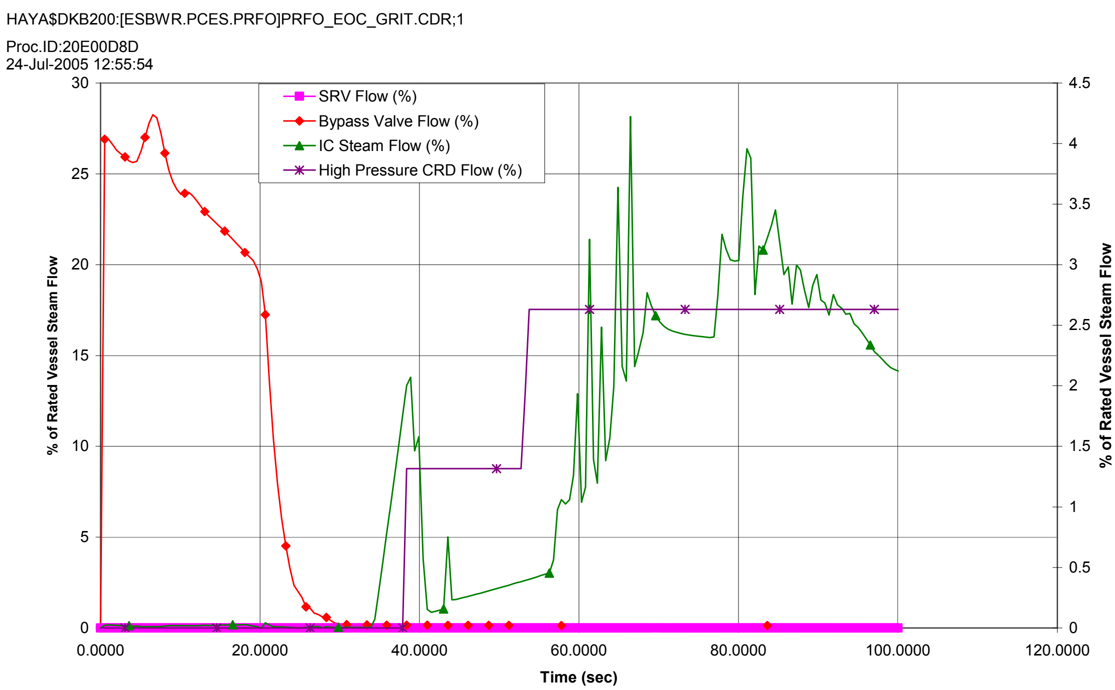


Figure 15.3-3b. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

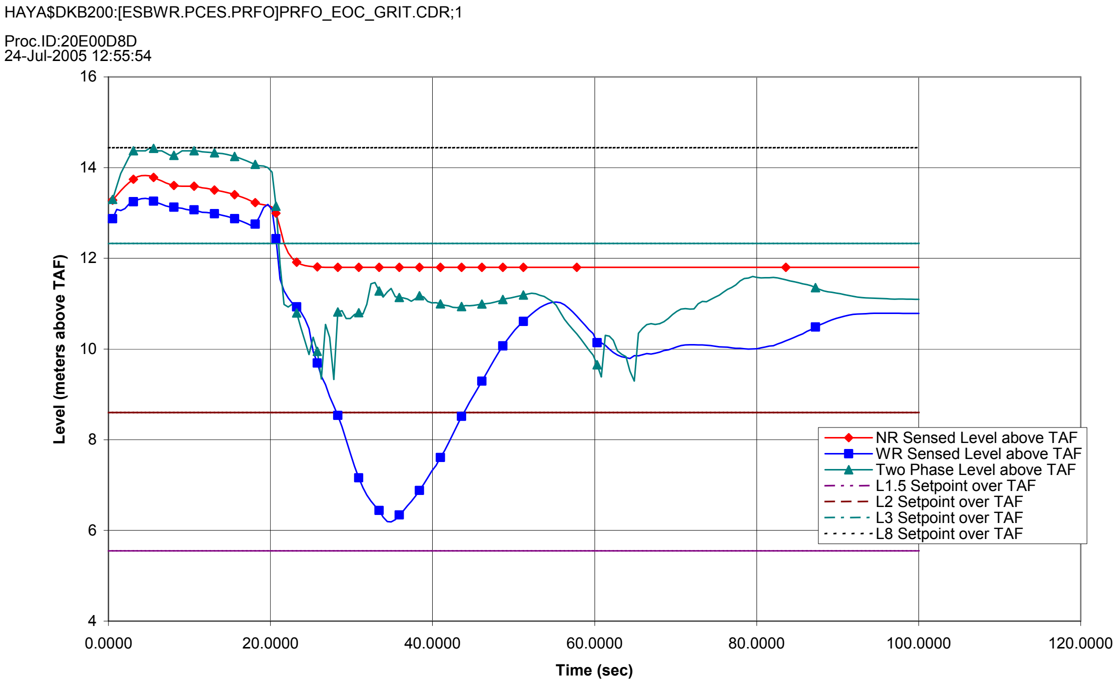


Figure 15.3-3c. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

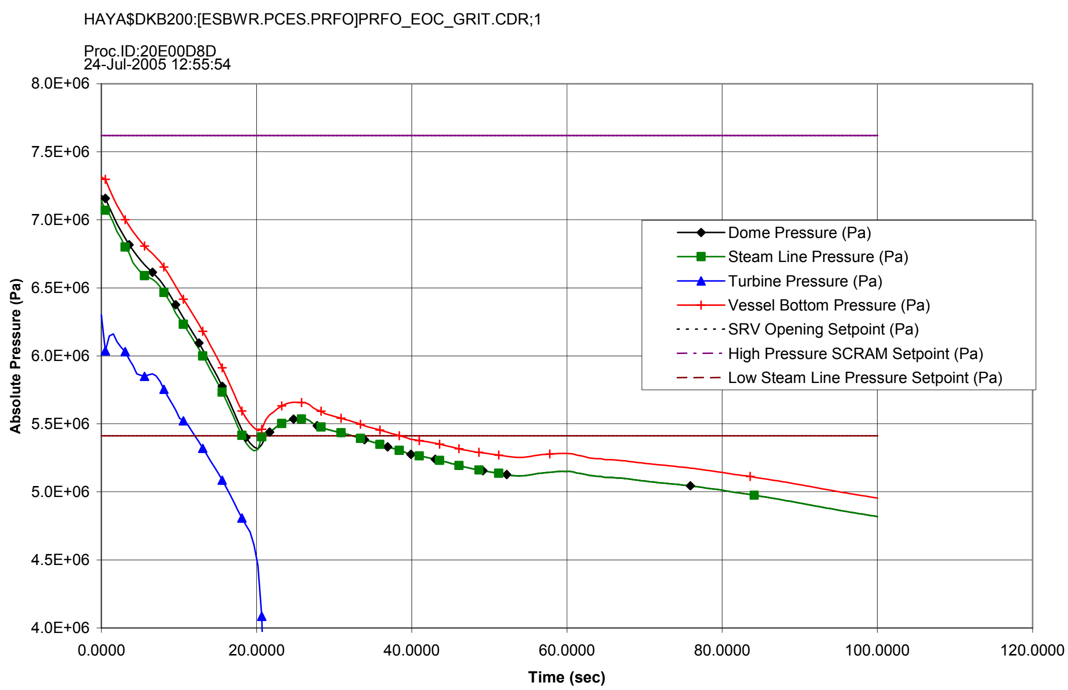


Figure 15.3-3d. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

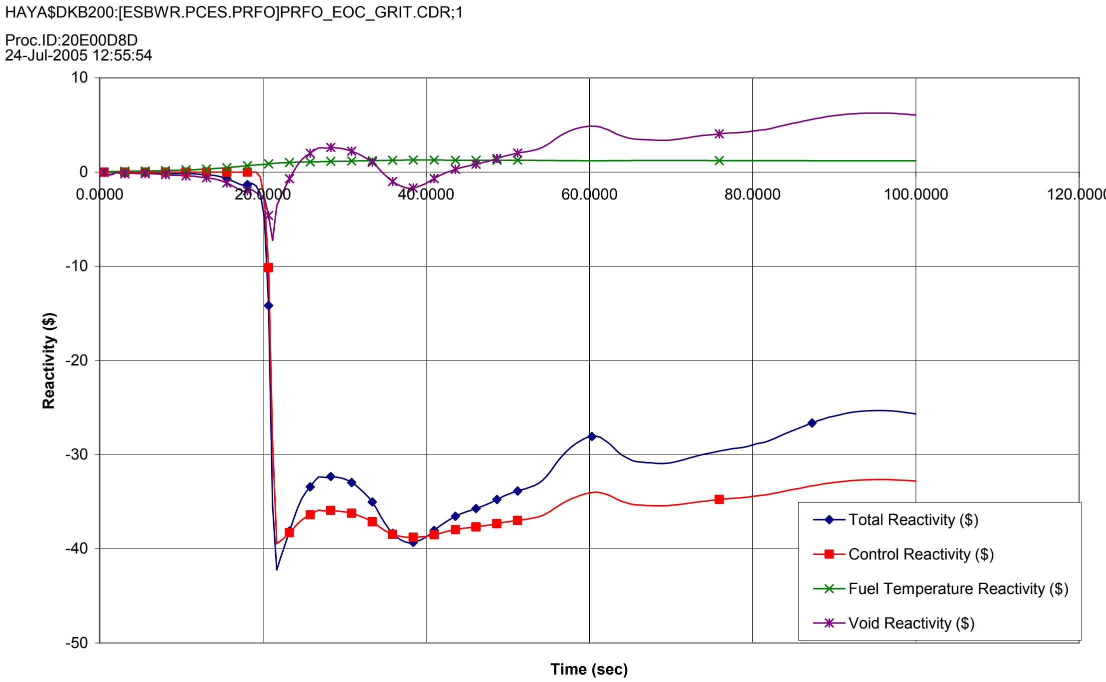


Figure 15.3-3e. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

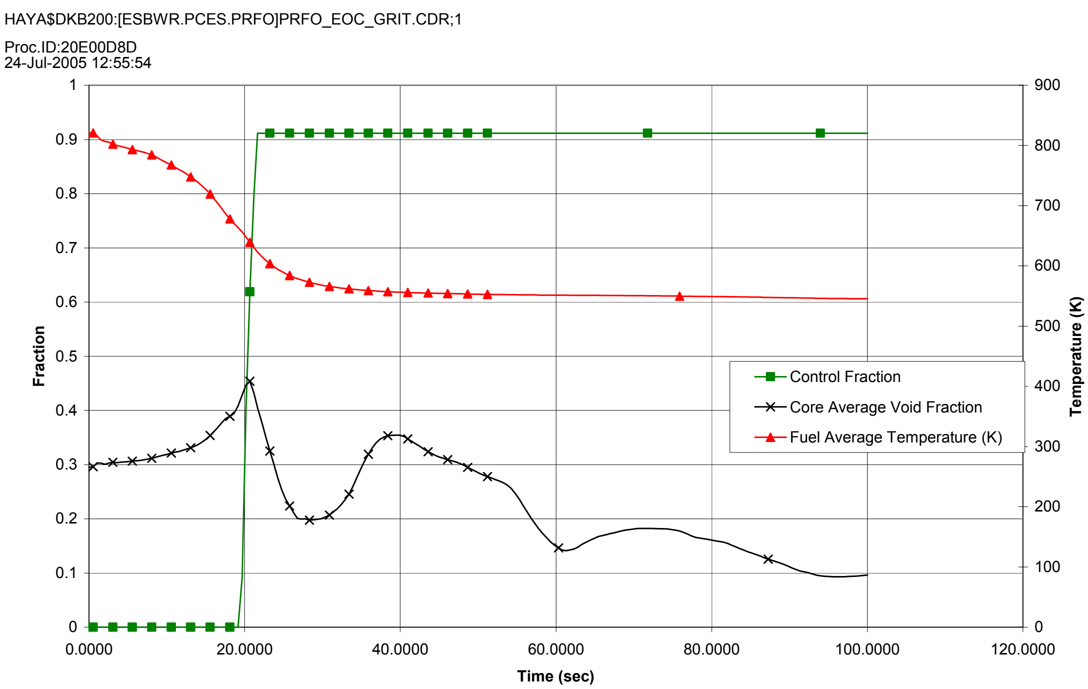


Figure 15.3-3f. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

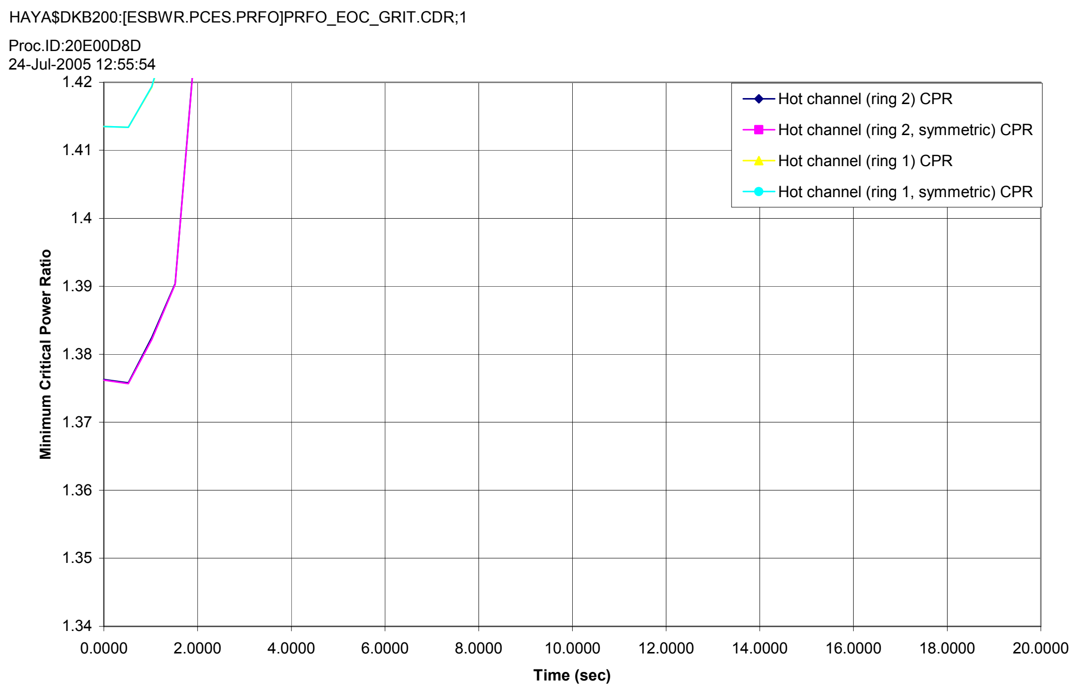


Figure 15.3-3g. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

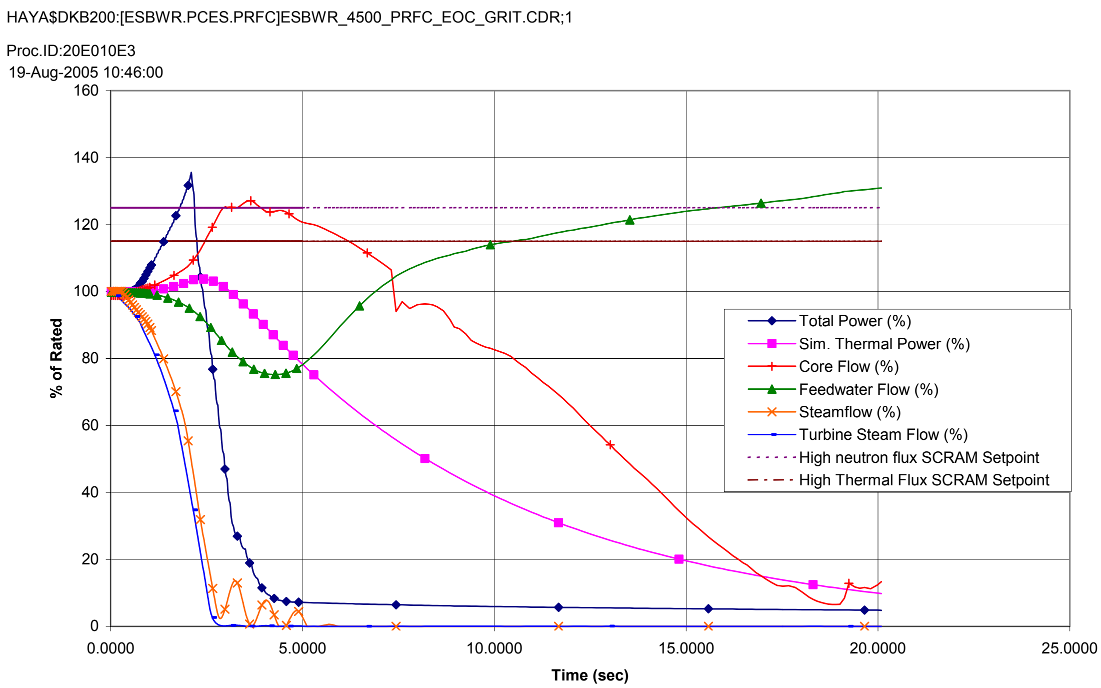


Figure 15.3-4a. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

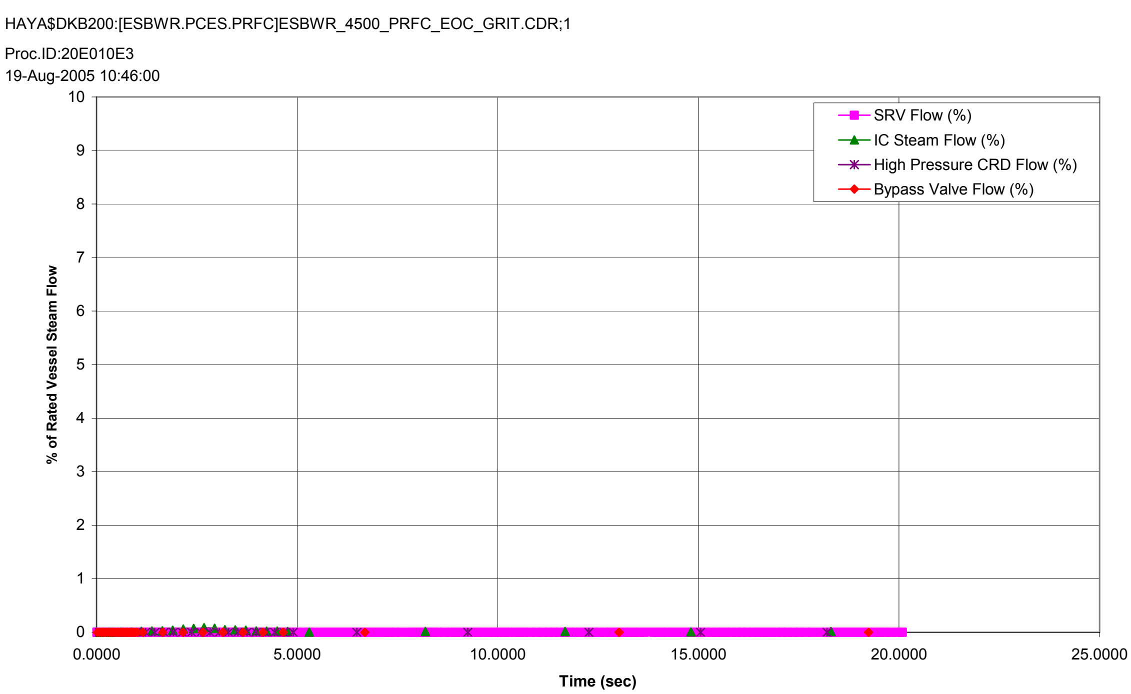


Figure 15.3-4b. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

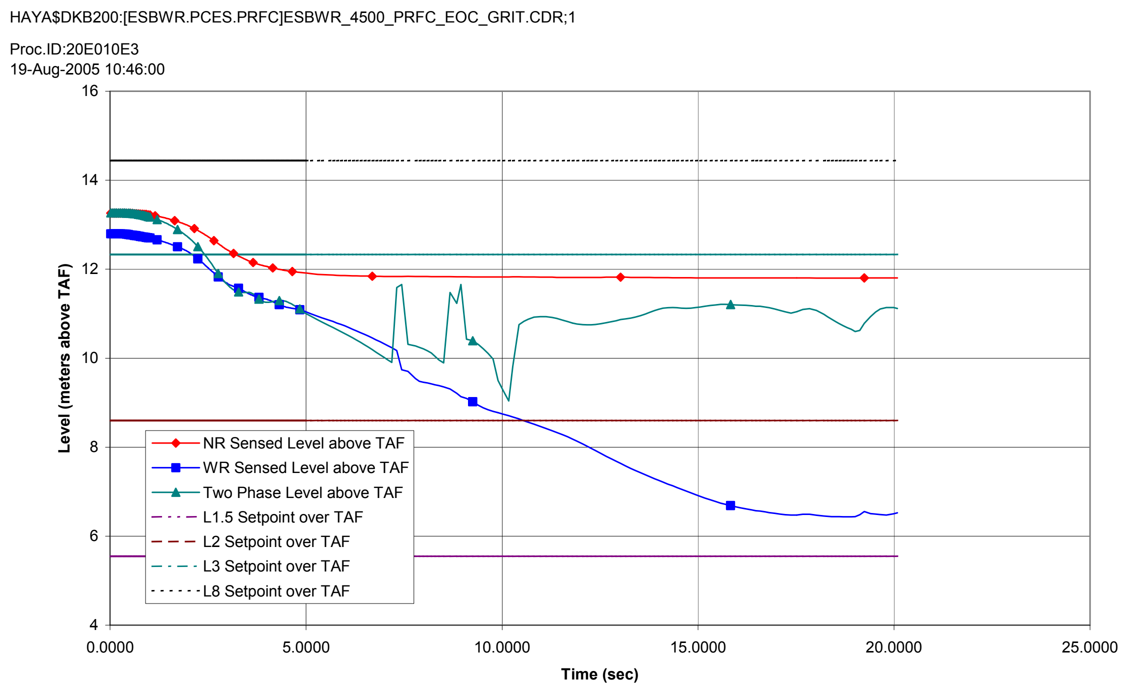


Figure 15.3-4c. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

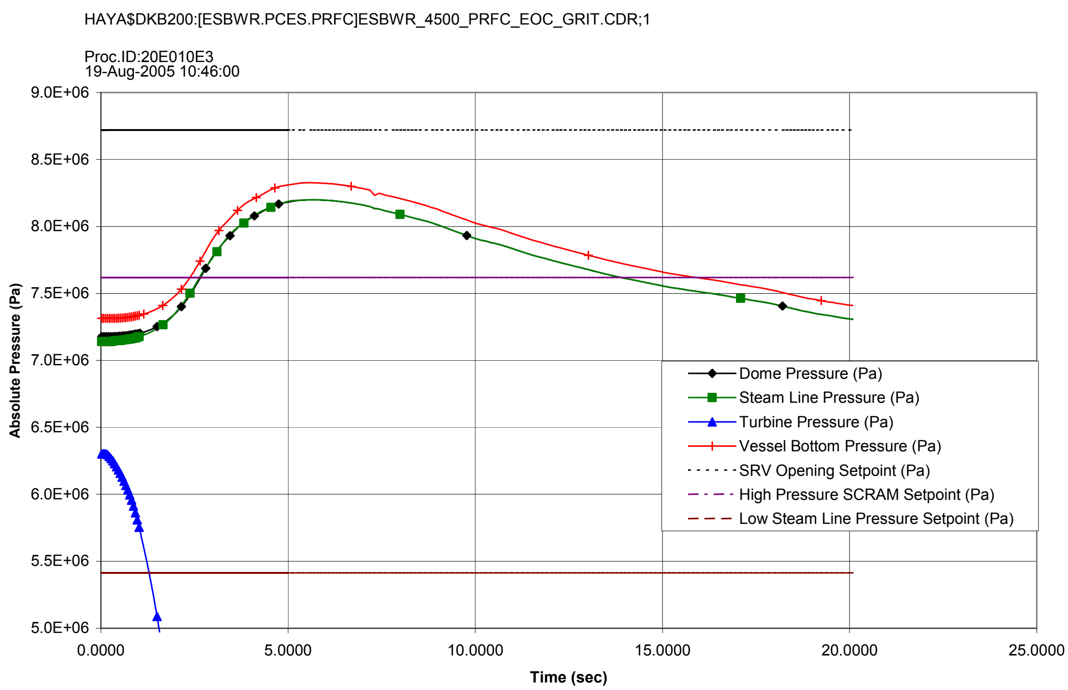


Figure 15.3-4d. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

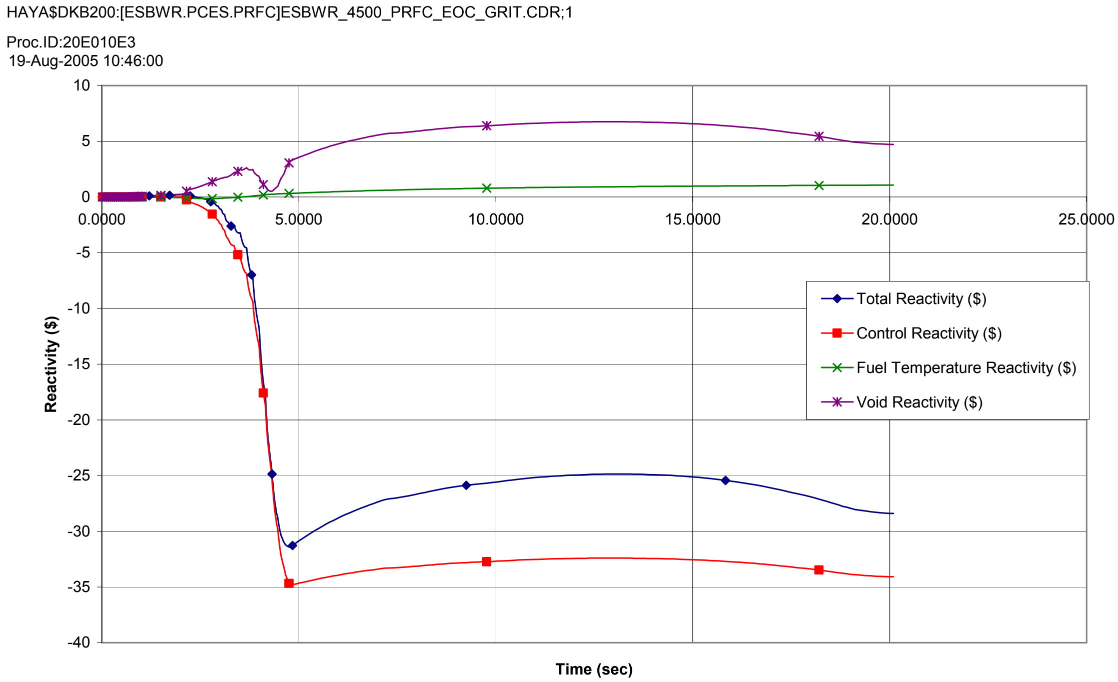


Figure 15.3-4e. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

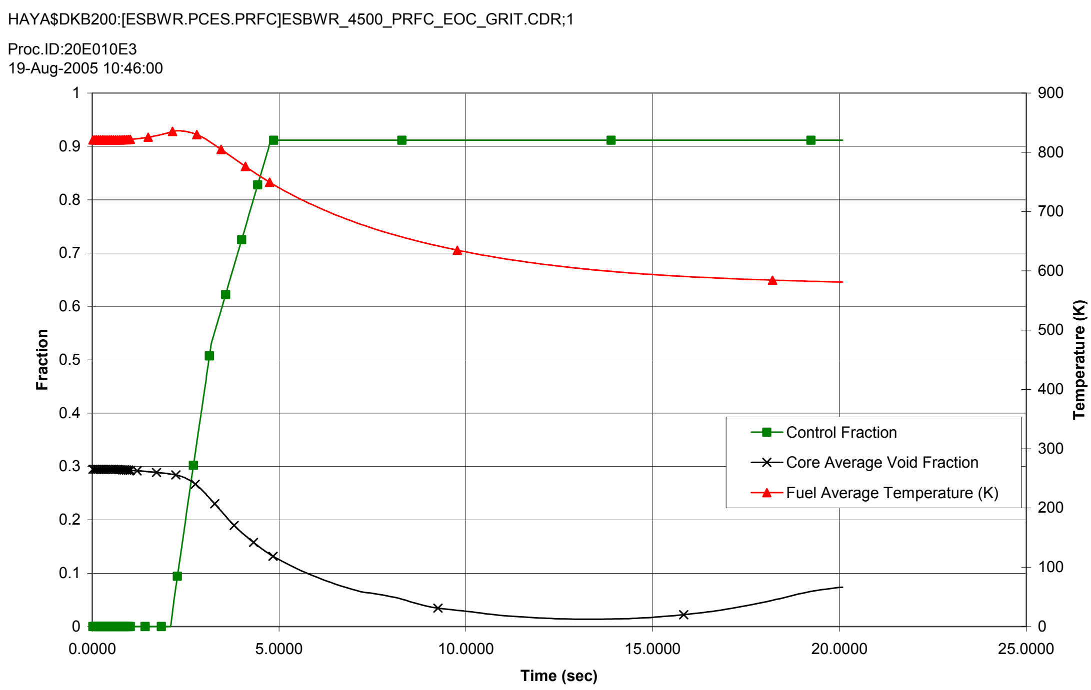


Figure 15.3-4f. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

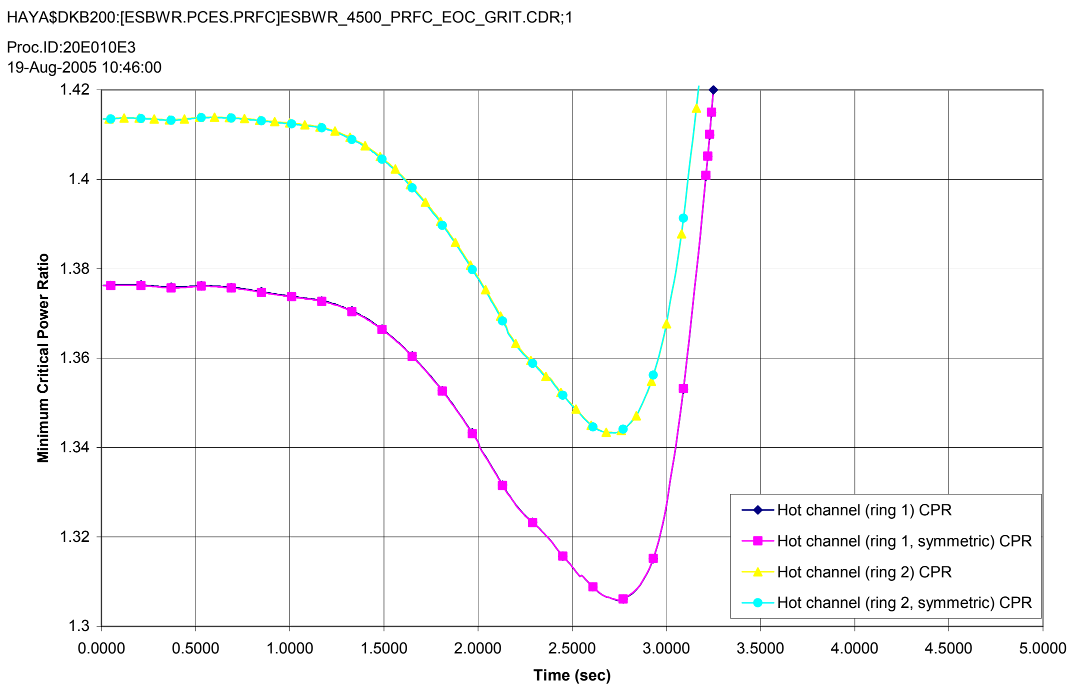


Figure 15.3-4g. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

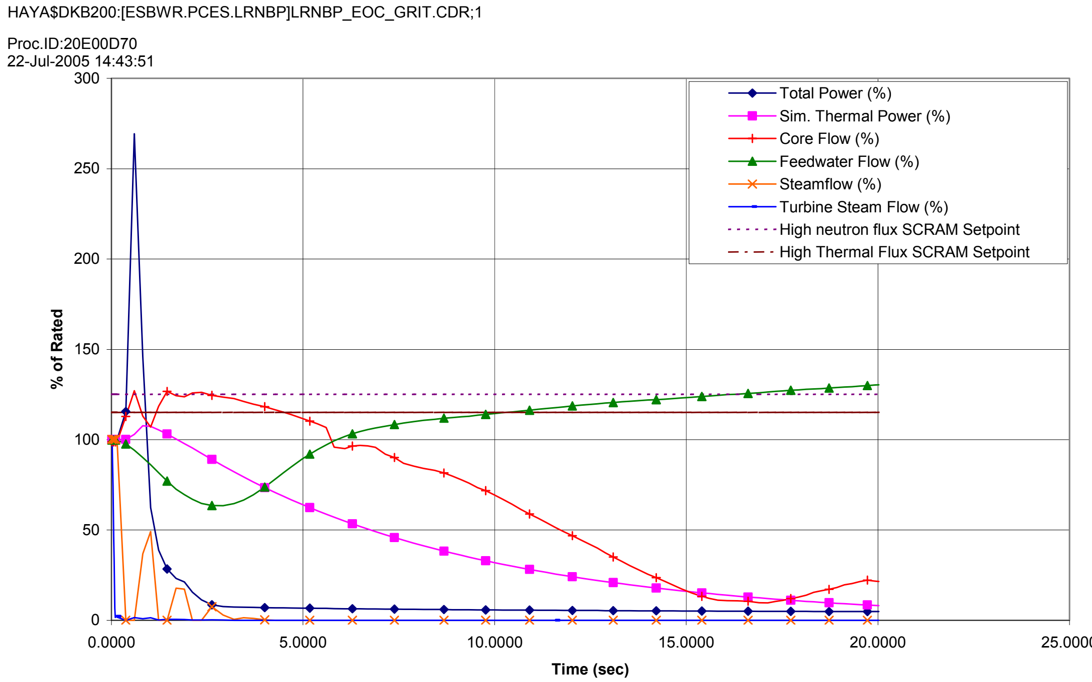


Figure 15.3-5a. Generator Load Rejection With Total Turbine Bypass Failure

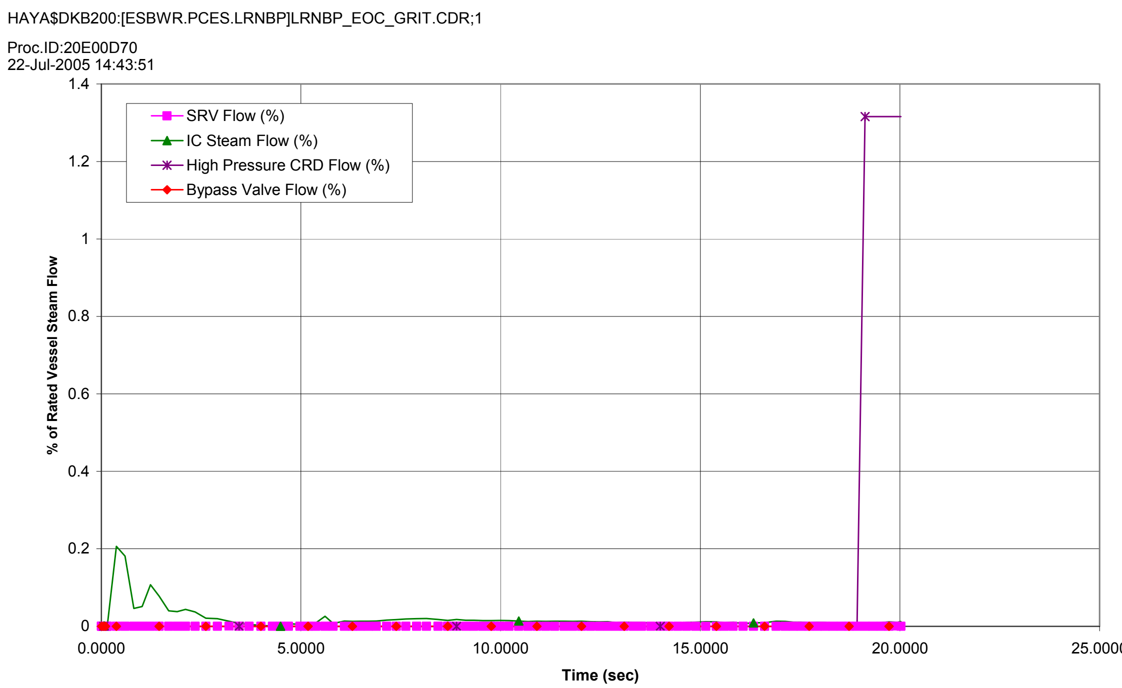


Figure 15.3-5b. Generator Load Rejection With Total Turbine Bypass Failure

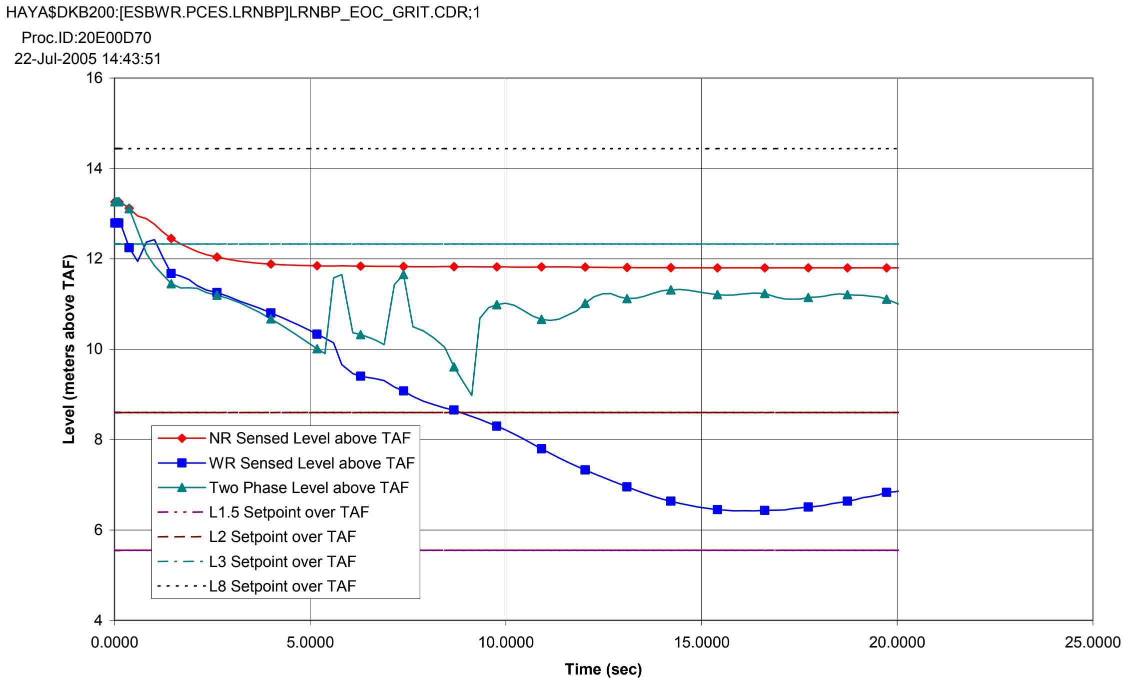


Figure 15.3-5c. Generator Load Rejection With Total Turbine Bypass Failure

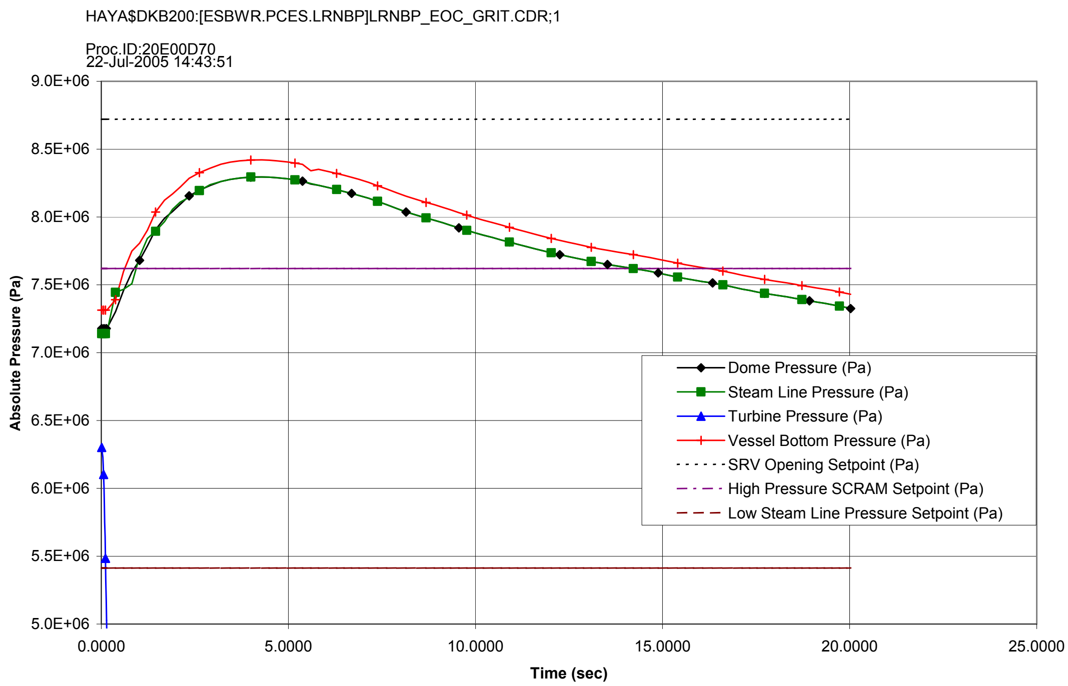


Figure 15.3-5d. Generator Load Rejection With Total Turbine Bypass Failure

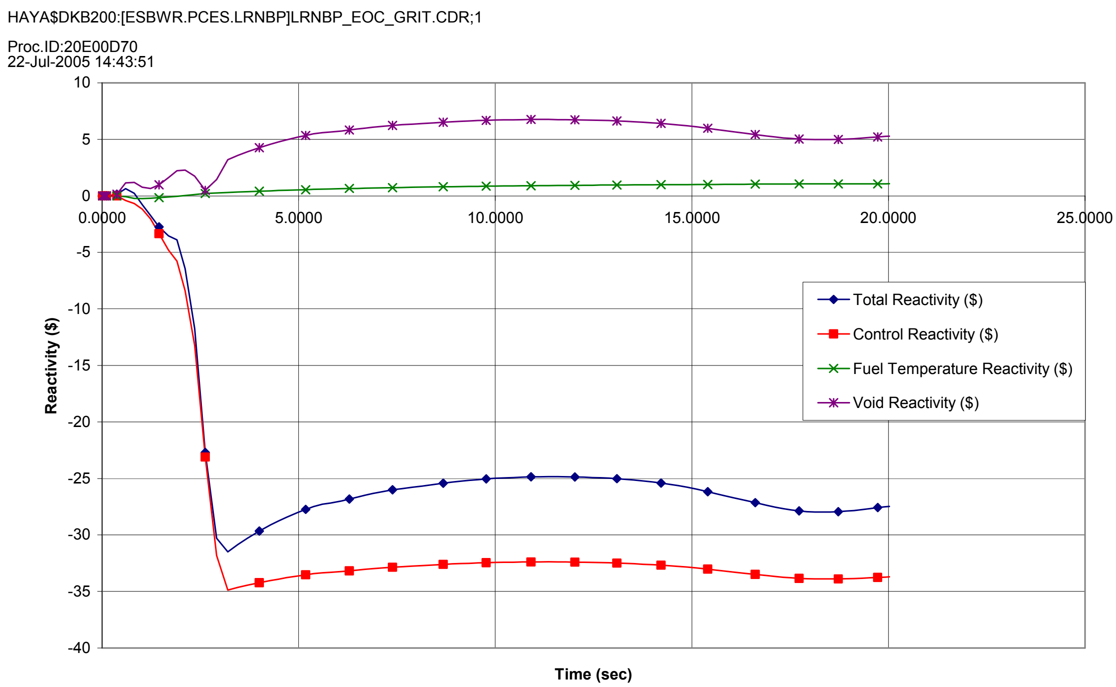


Figure 15.3-5e. Generator Load Rejection With Total Turbine Bypass Failure

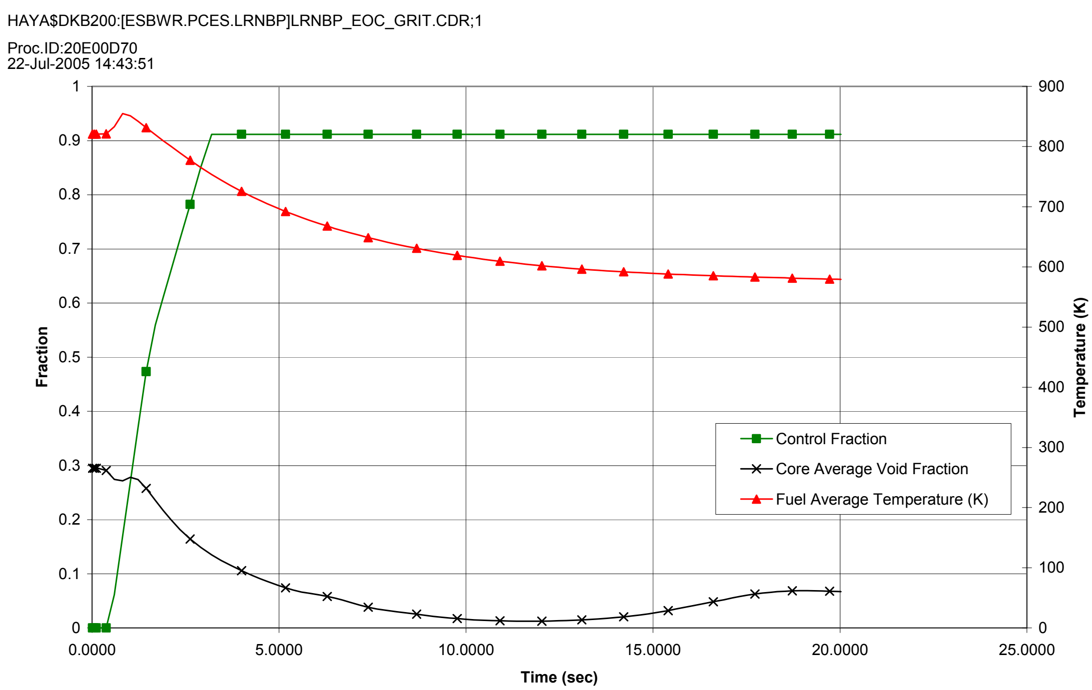


Figure 15.3-5f. Generator Load Rejection With Total Turbine Bypass Failure

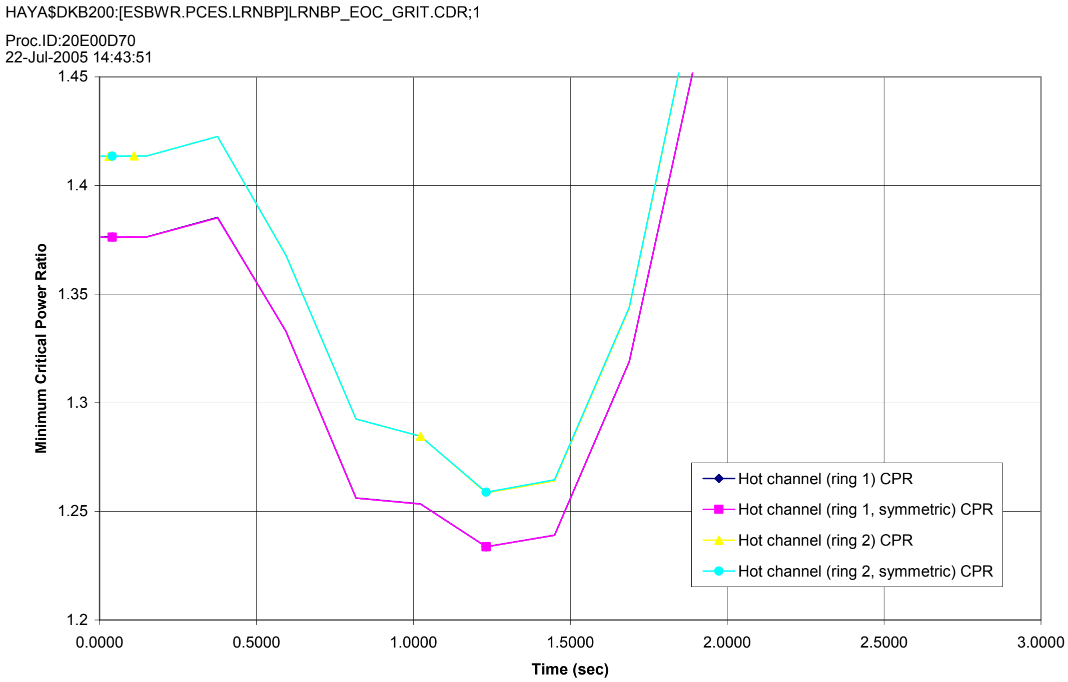


Figure 15.3-5g. Generator Load Rejection With Total Turbine Bypass Failure

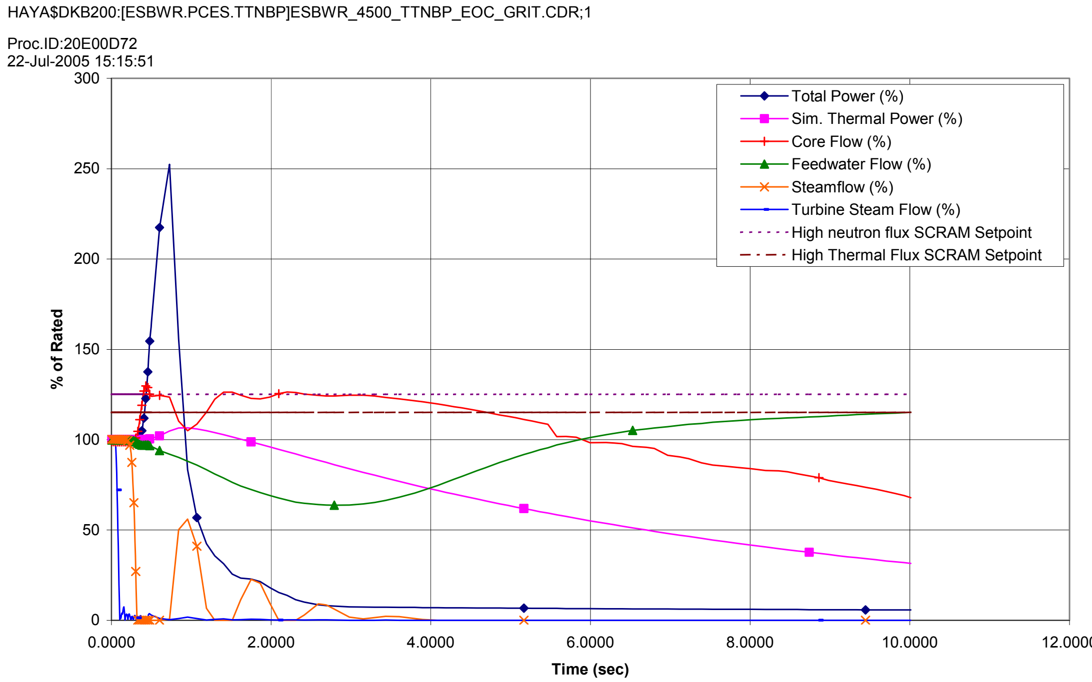


Figure 15.3-6a. Turbine Trip With Total Turbine Bypass Failure

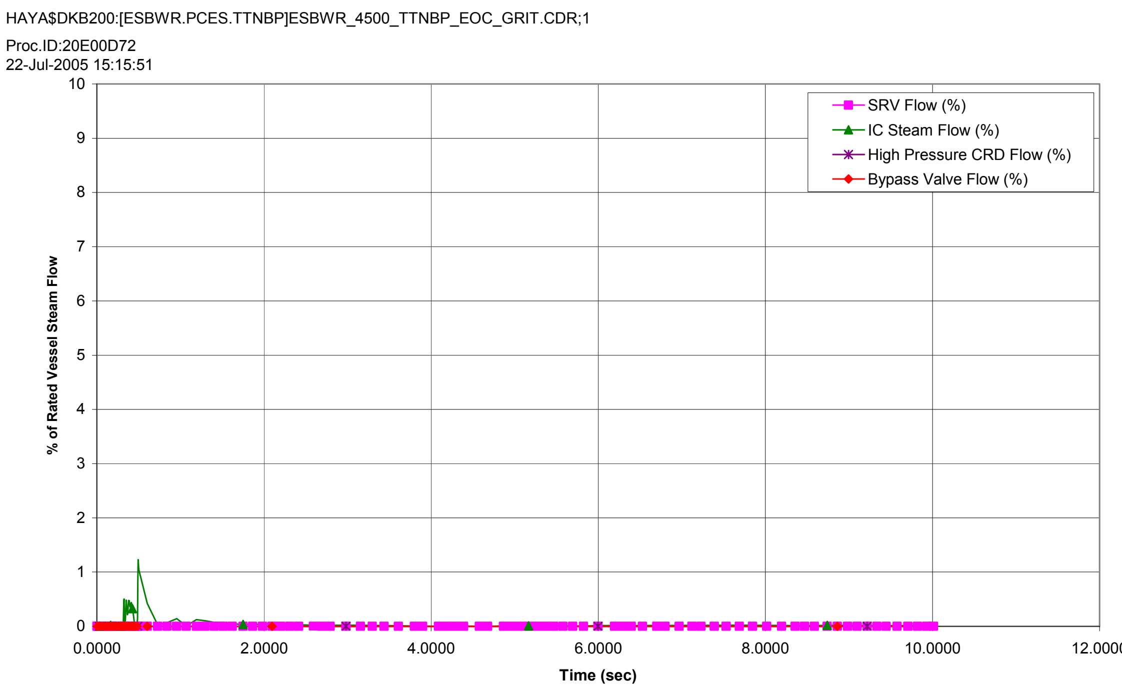


Figure 15.3-6b. Turbine Trip With Total Turbine Bypass Failure

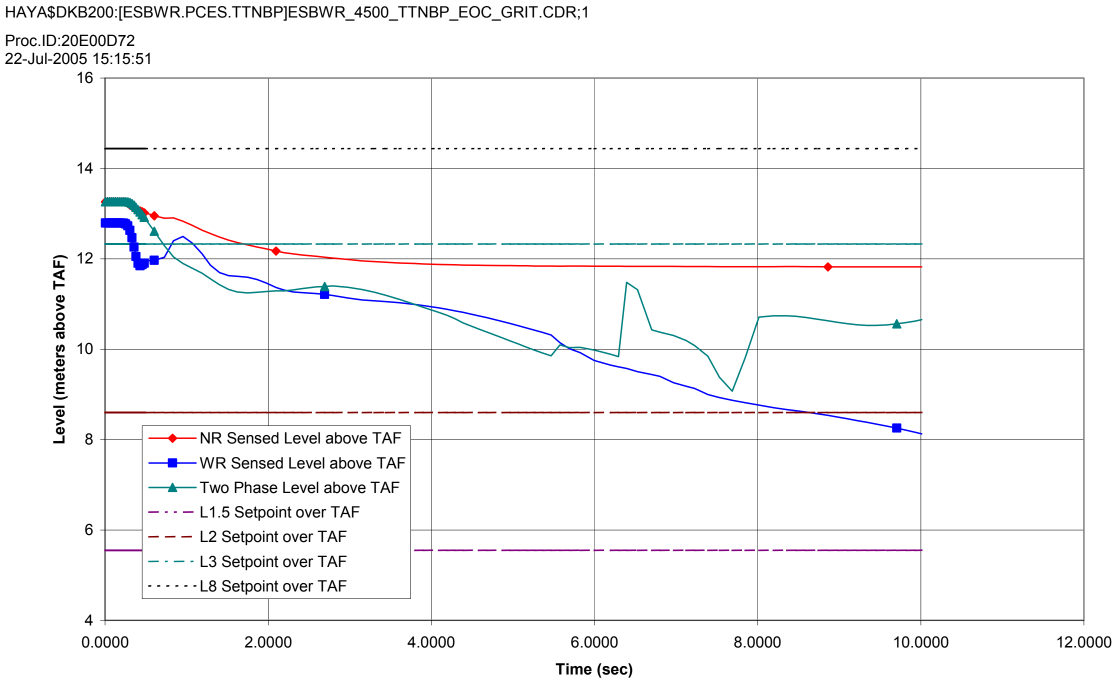


Figure 15.3-6c. Turbine Trip With Total Turbine Bypass Failure

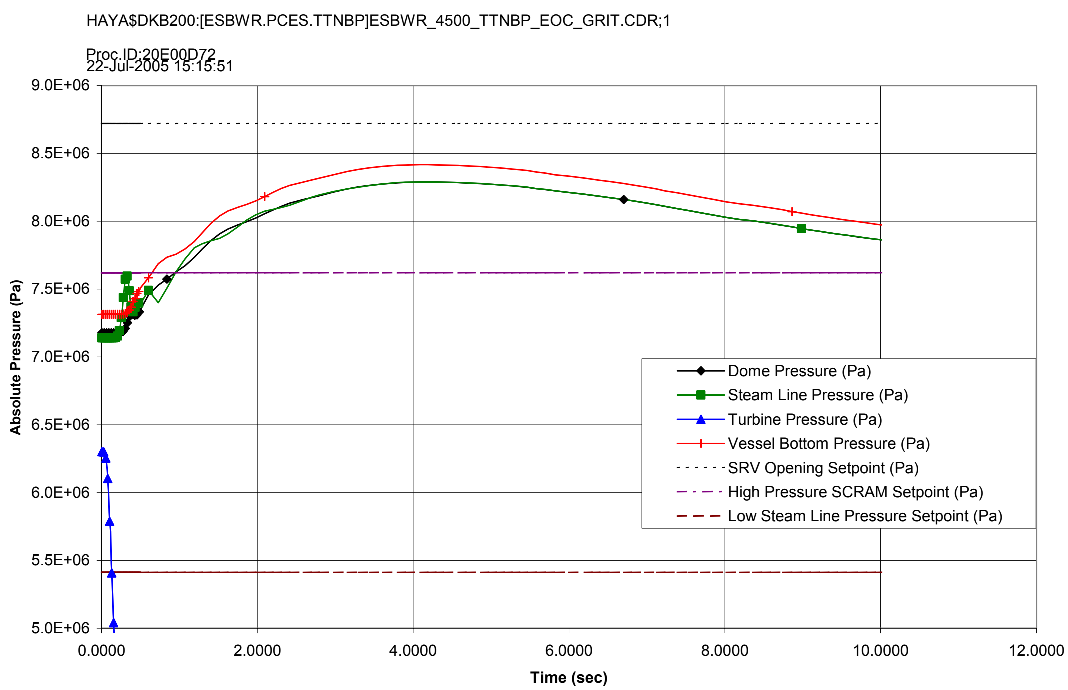


Figure 15.3-6d. Turbine Trip With Total Turbine Bypass Failure

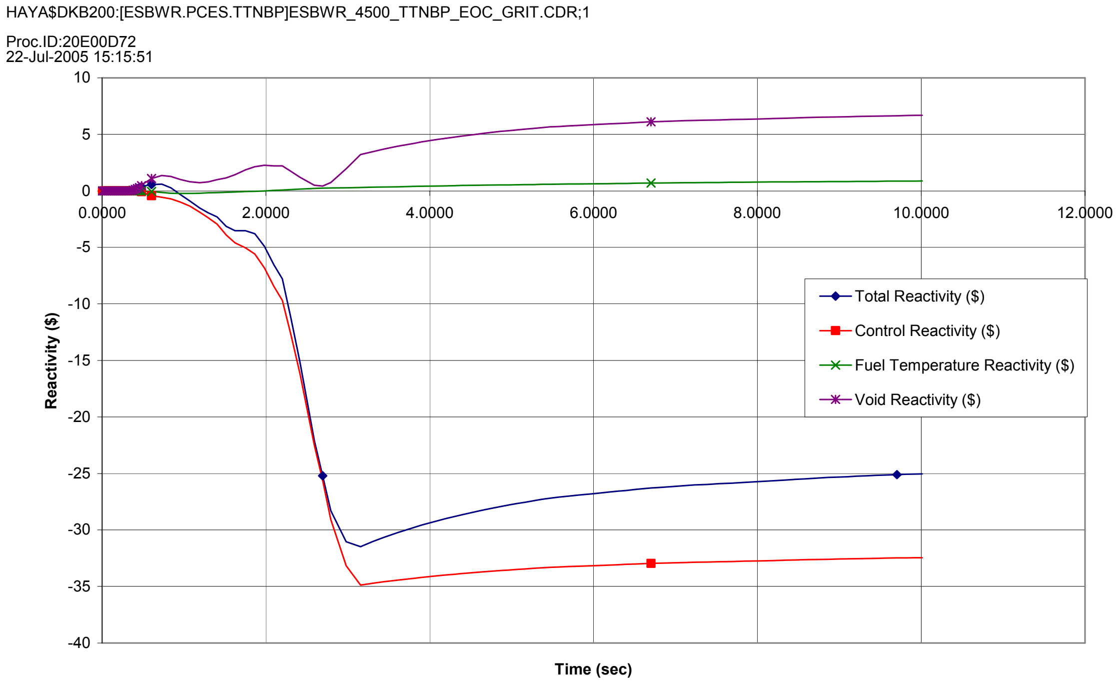


Figure 15.3-6e. Turbine Trip With Total Turbine Bypass Failure

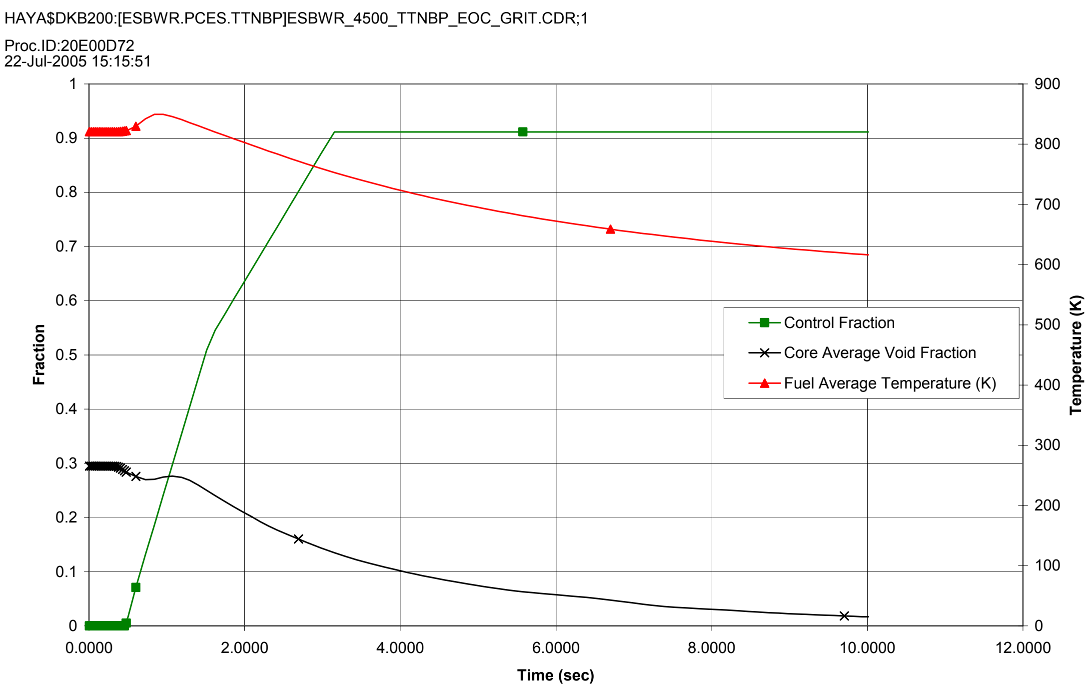


Figure 15.3-6f. Turbine Trip With Total Turbine Bypass Failure

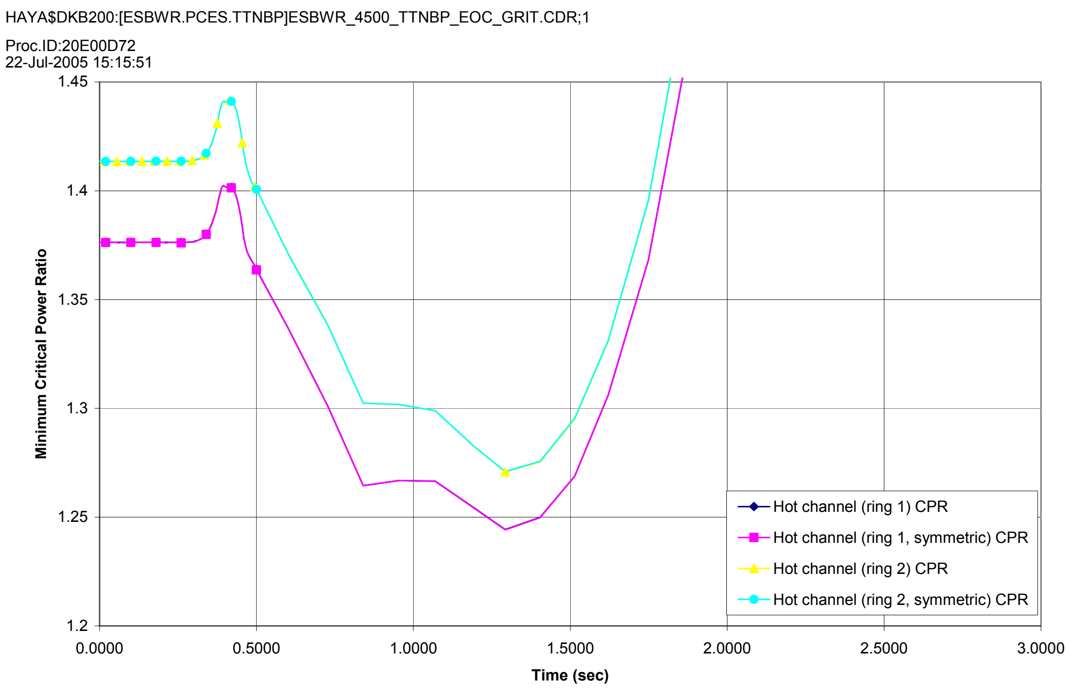


Figure 15.3-6g. Turbine Trip With Total Turbine Bypass Failure

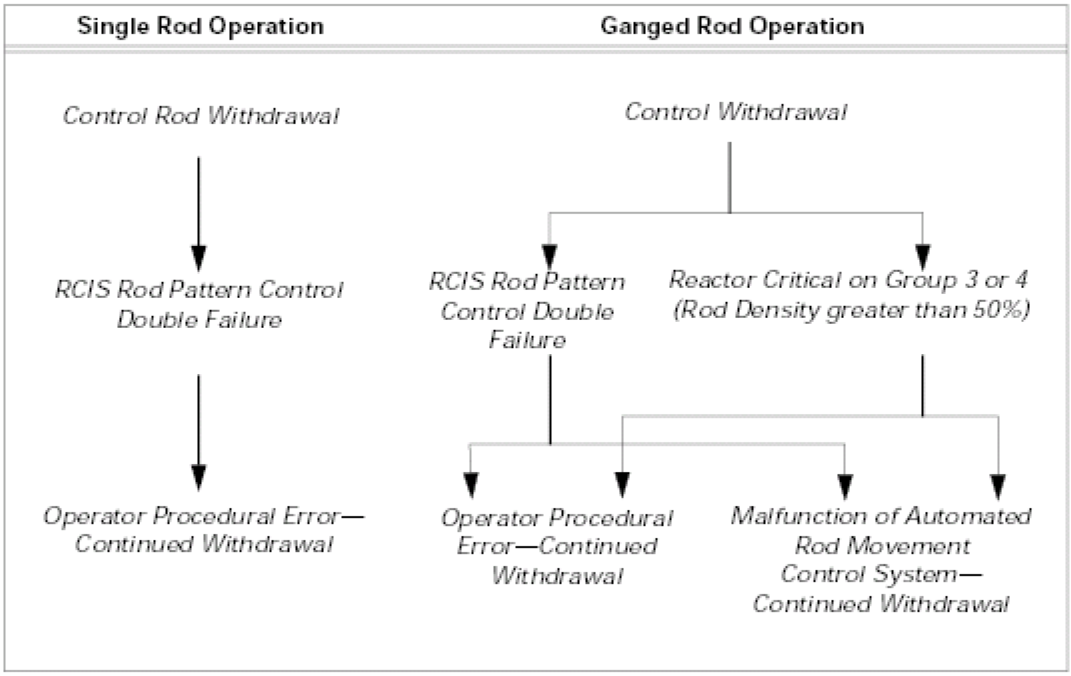


Figure 15.3-7a. Causes of Control Rod Withdrawal Error

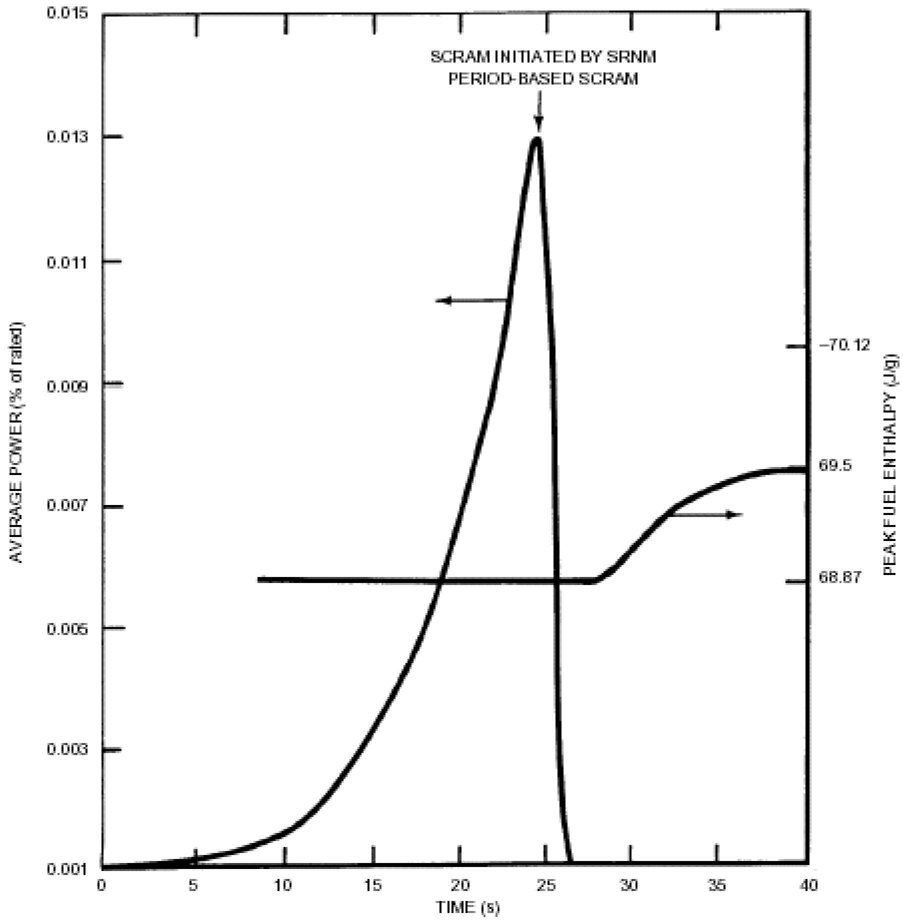


Figure 15.3-7b. Transient Changes for Control Rod Withdrawal Error During Startup (Representative BWR Analysis)

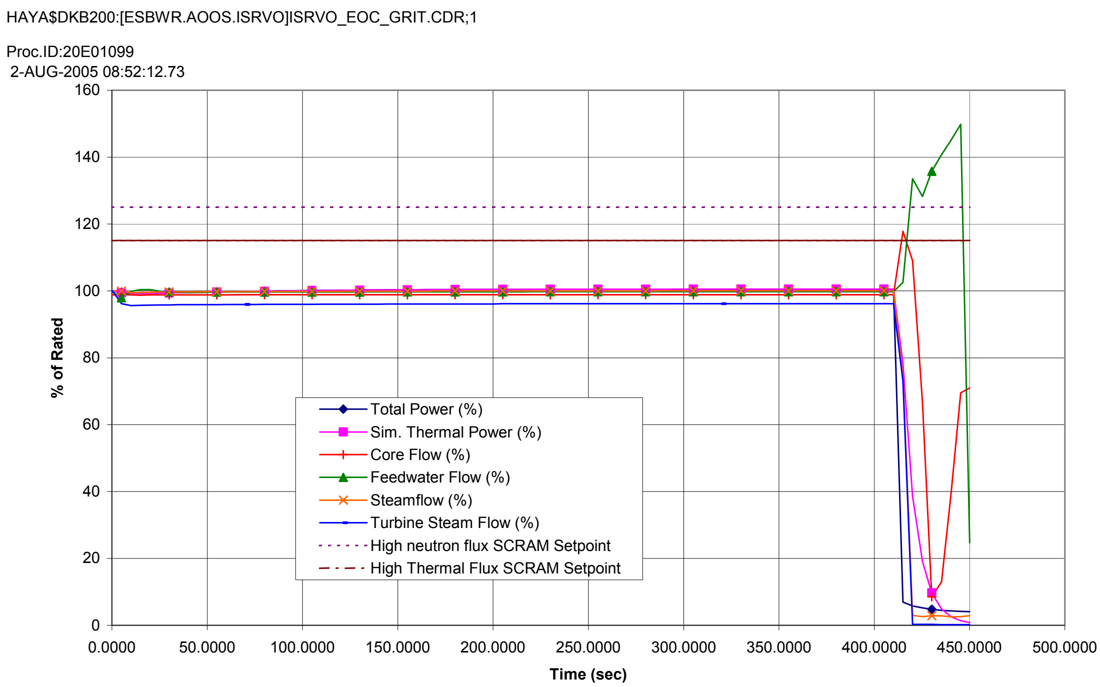


Figure 15.3-8a. Inadvertent SRV opening

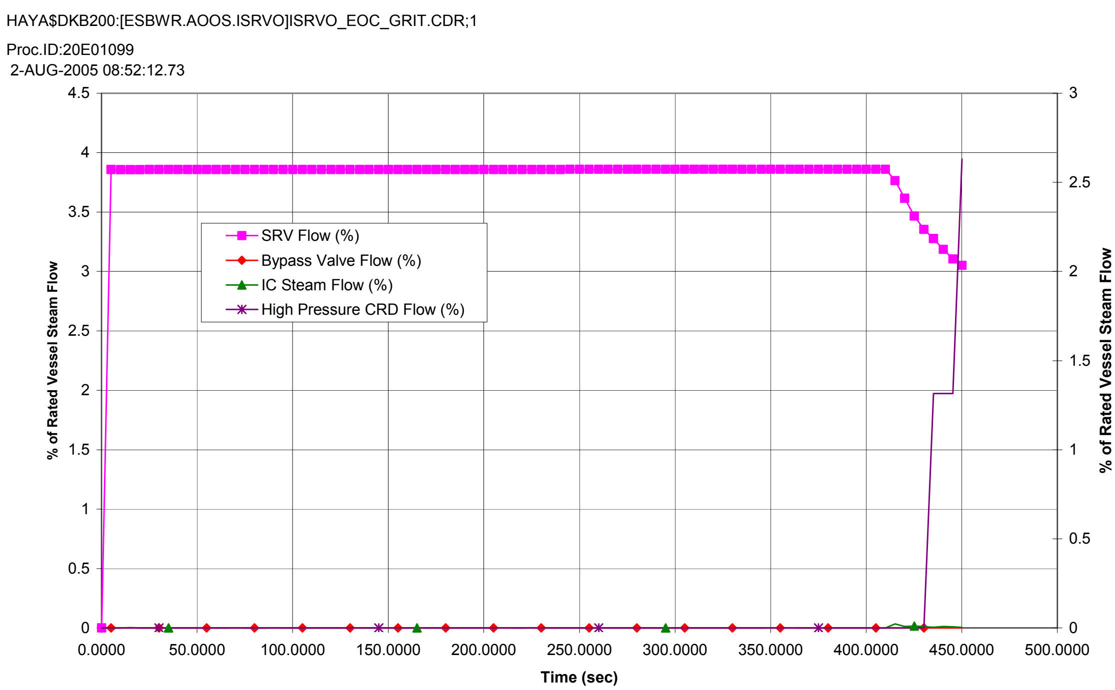


Figure 15.3-8b. Inadvertent SRV opening

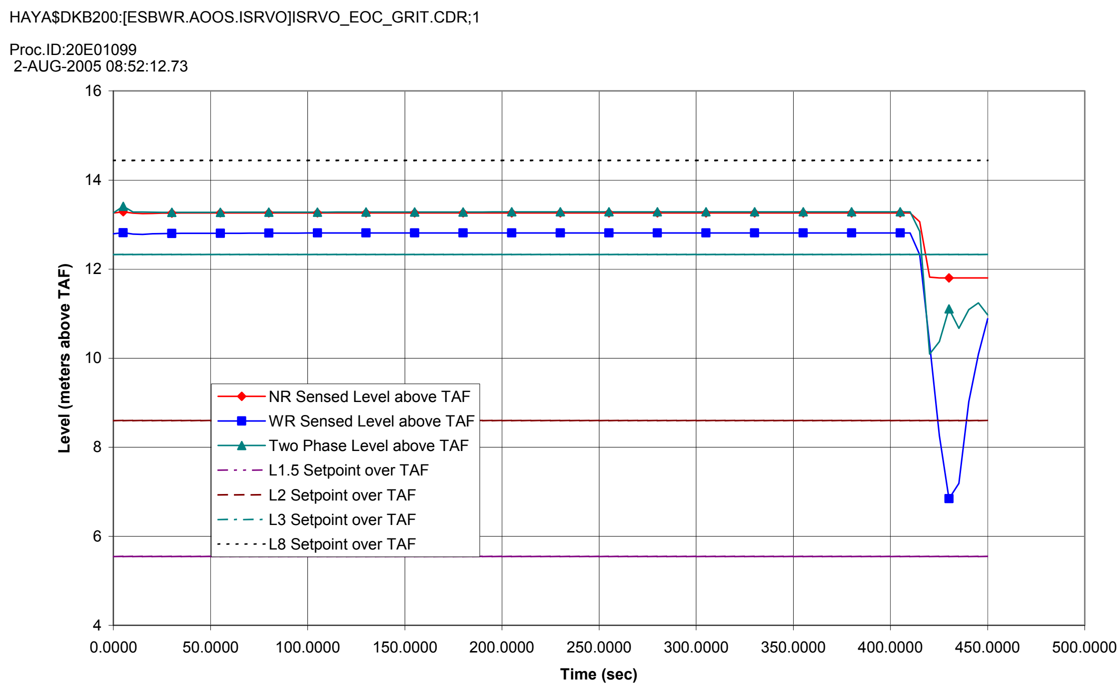


Figure 15.3-8c. Inadvertent SRV opening

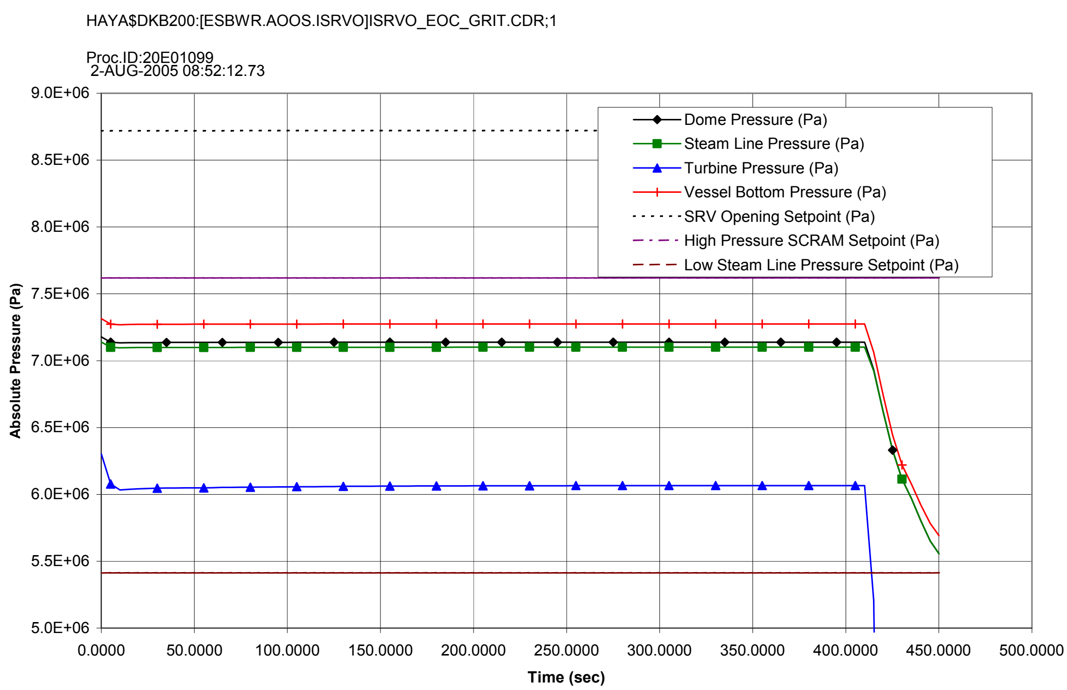


Figure 15.3-8d. Inadvertent SRV opening

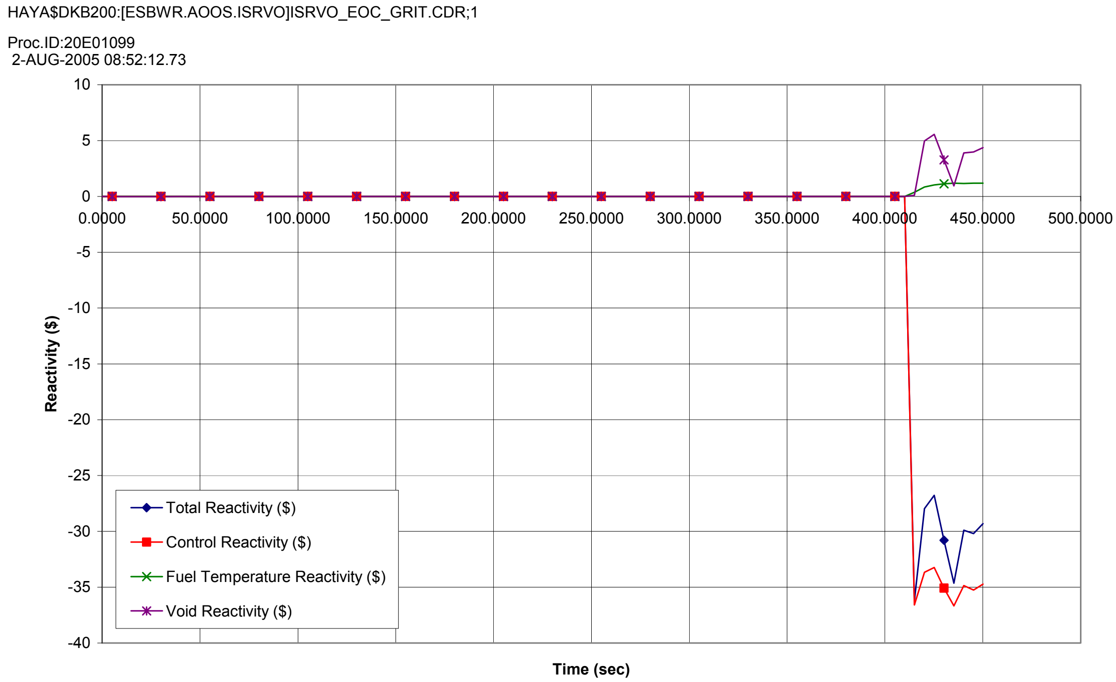


Figure 15.3-8e. Inadvertent SRV opening

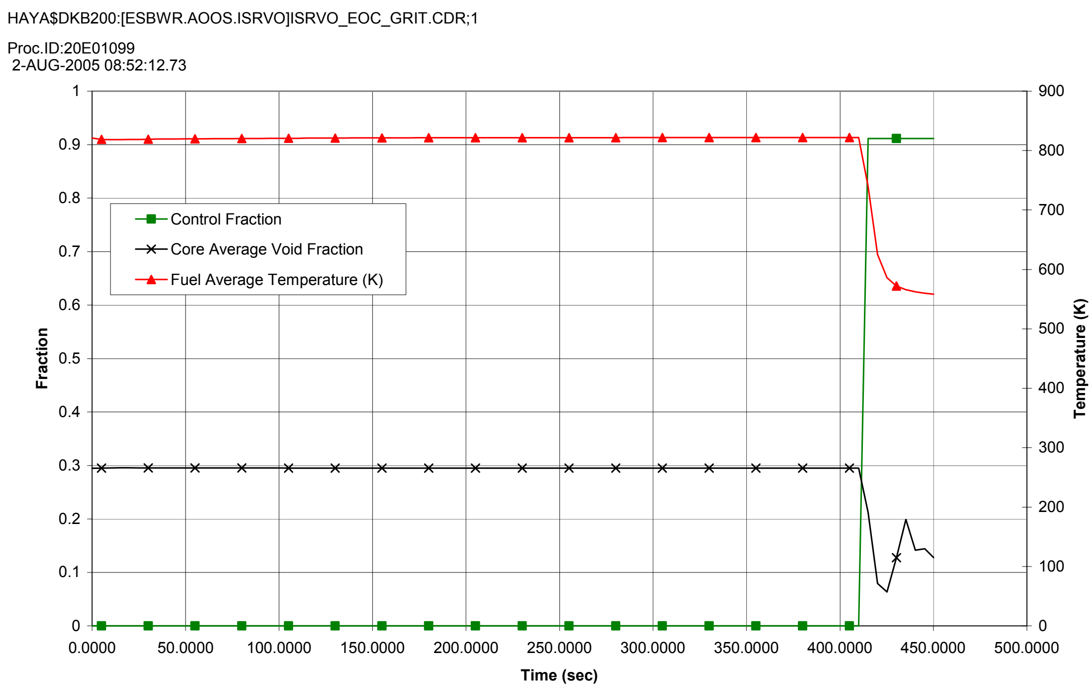


Figure 15.3-8f. Inadvertent SRV opening

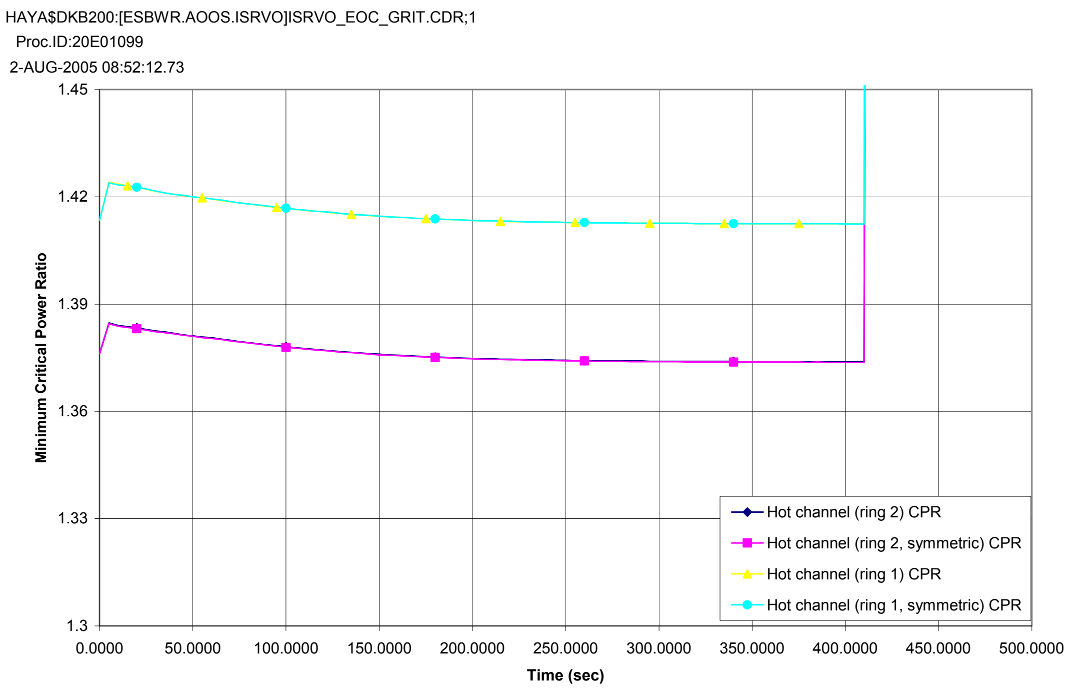


Figure 15.3-8g. Inadvertent SRV opening

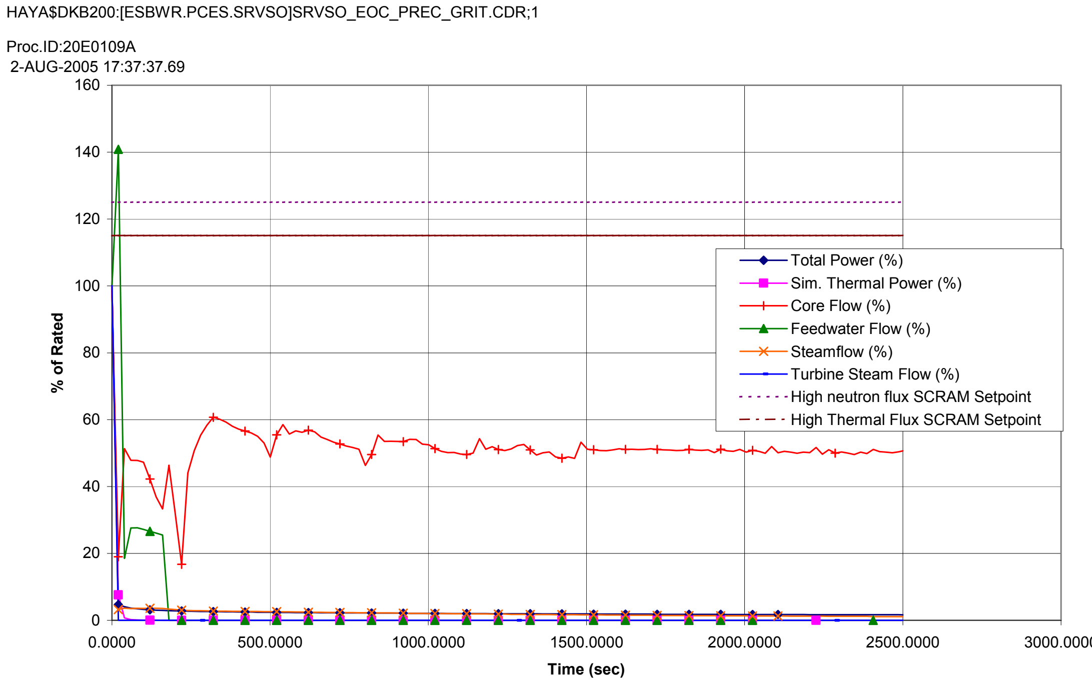


Figure 15.3-9a. Stuck Open Safety Relief Valve

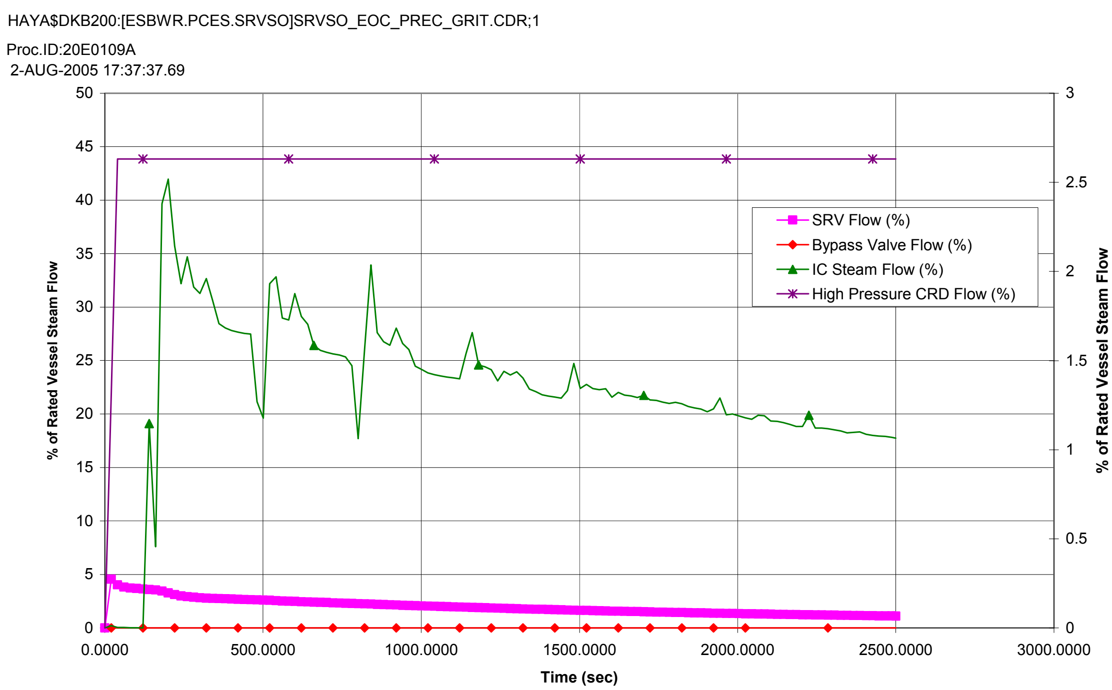


Figure 15.3-9b. Stuck Open Safety Relief Valve

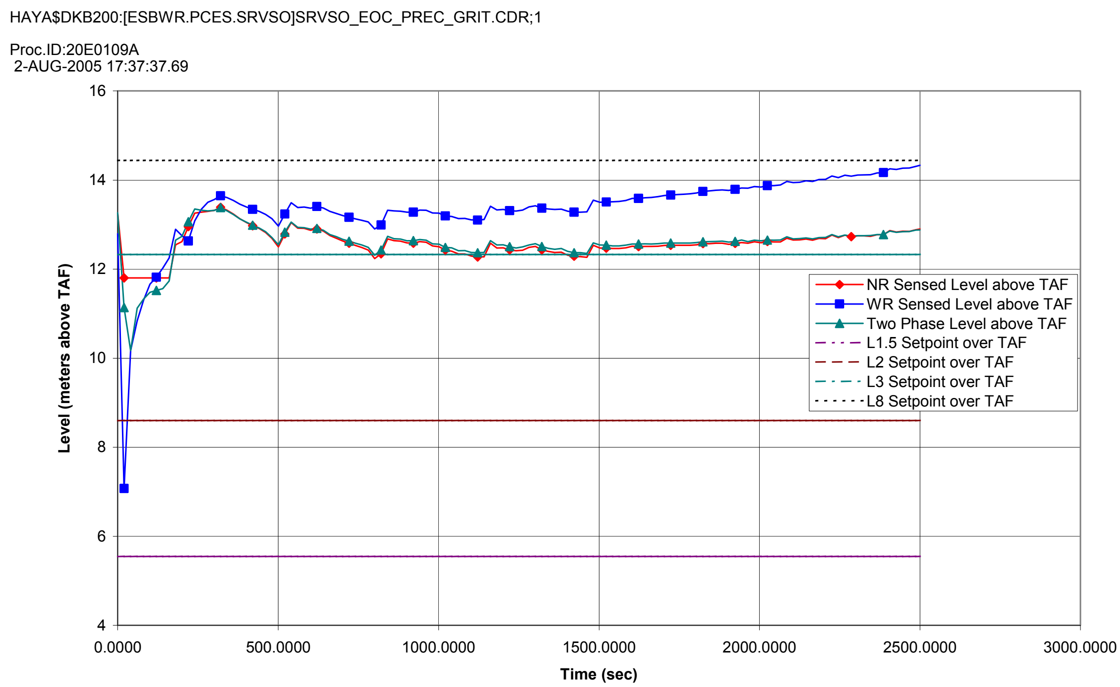


Figure 15.3-9c. Stuck Open Safety Relief Valve

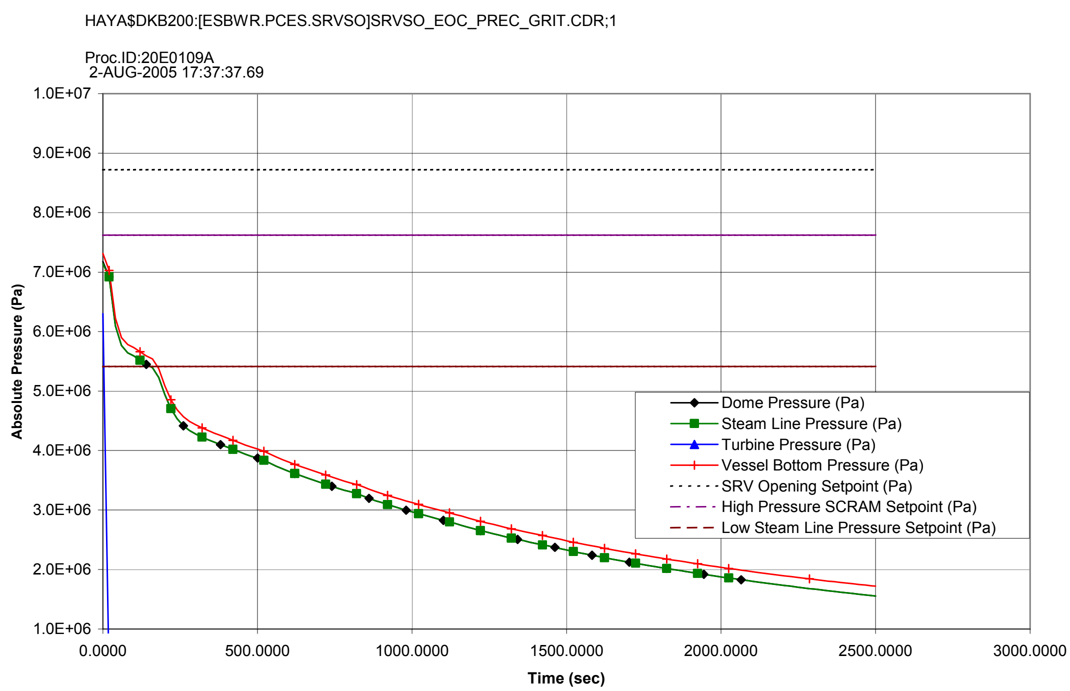


Figure 15.3-9d. Stuck Open Safety Relief Valve

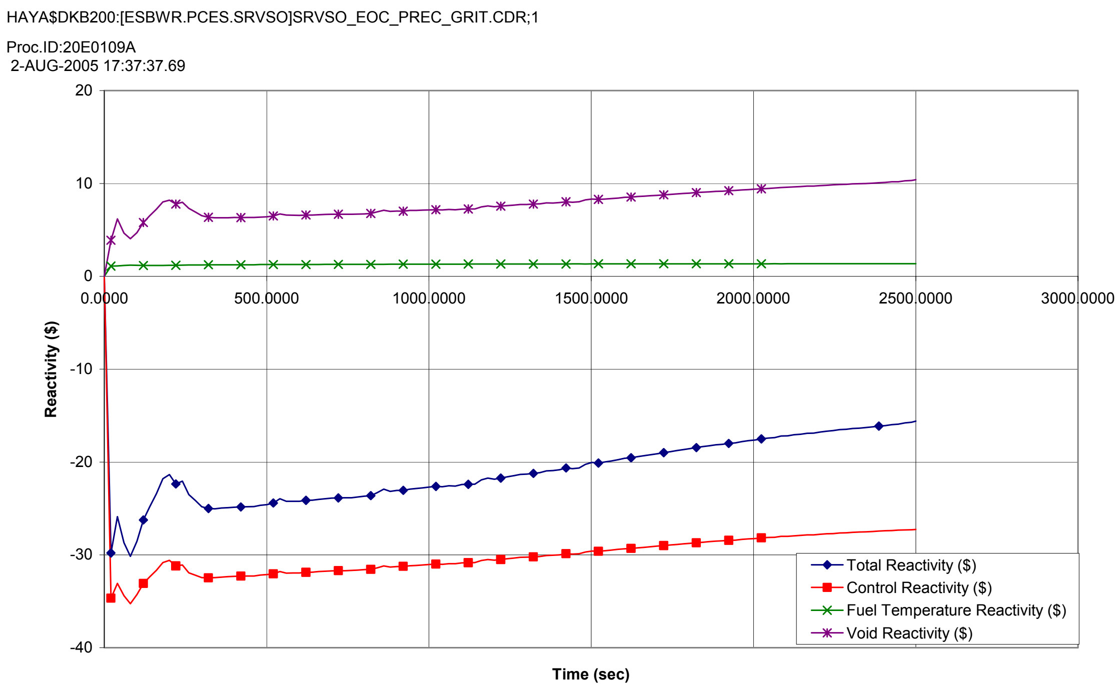


Figure 15.3-9e. Stuck Open Safety Relief Valve

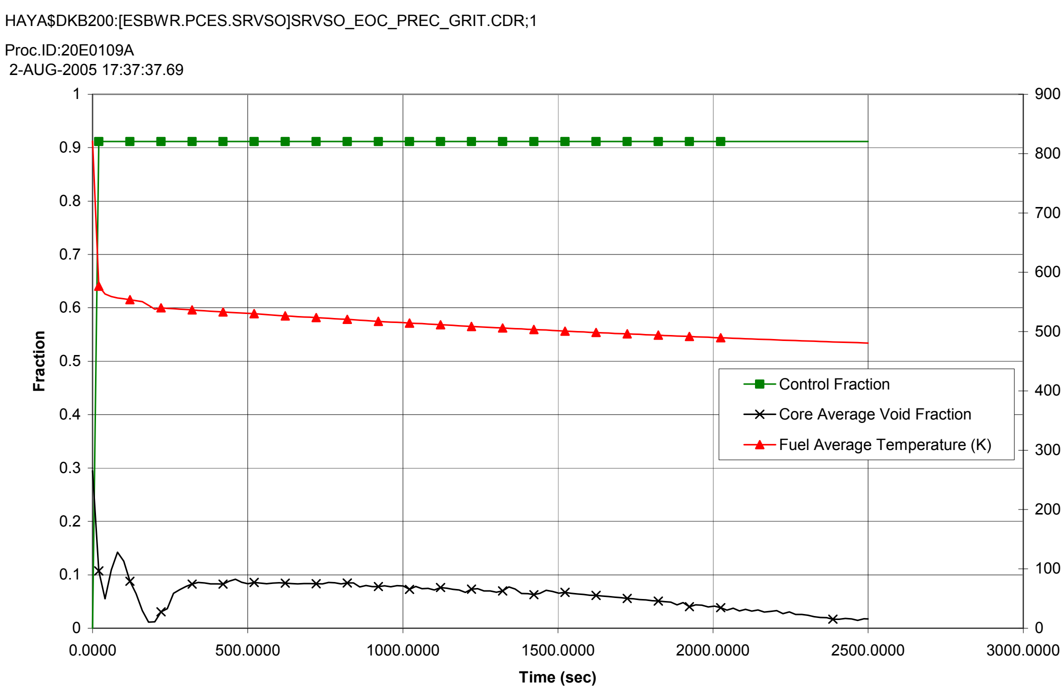


Figure 15.3-9f. Stuck Open Safety Relief Valve

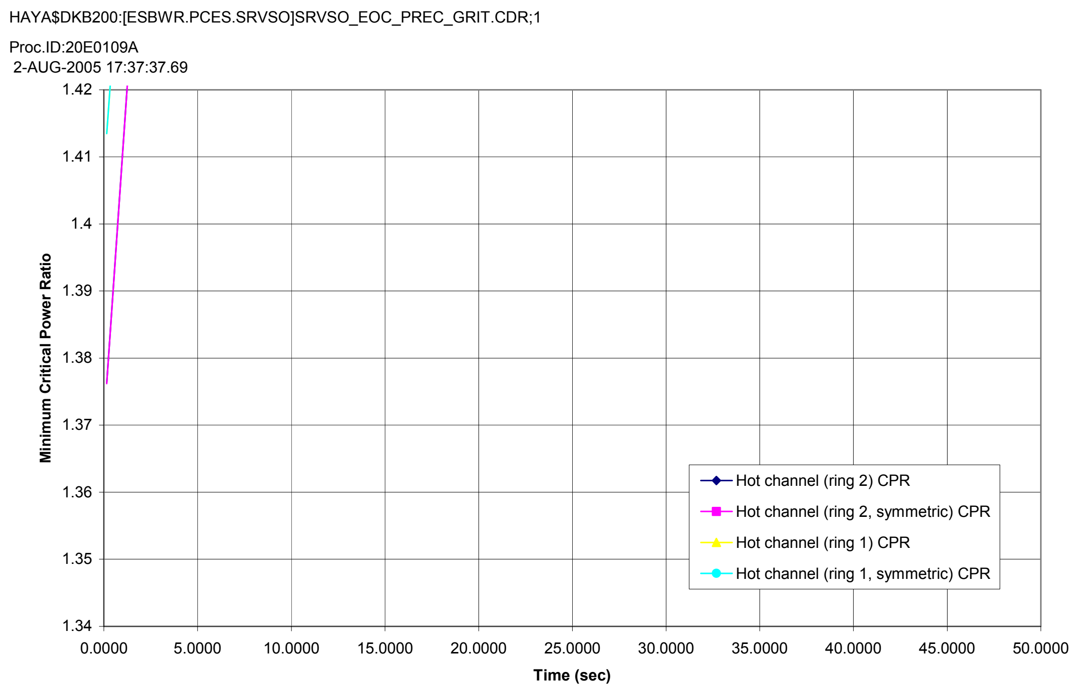


Figure 15.3-9g. Stuck Open Safety Relief Valve

15.4 ANALYSIS OF ACCIDENTS

15.4.1 Fuel Handling Accident

15.4.1.1 Identification of Causes

The fuel-handling accident is assumed to occur as a result of a failure of the fuel assembly lifting mechanism, resulting in dropping a raised fuel assembly onto the reactor core.

15.4.1.2 Sequence of Events and Systems Operation

Sequence of Events

The sequence of events is provided in Table 15.4-1.

Identification of Operator Actions

The following actions are carried out:

- initiate the evacuation of the Reactor Building fuel handling area and the locking of the fuel building doors;
- the fuel-handling foreman gives instructions to go immediately to the radiation protection decontamination area;
- the fuel-handling foreman makes the operations shift engineer aware of the accident;
- the shift engineer determines if the normal ventilation system has isolated;
- the shift engineer initiates action to determine the extent of potential radiation doses by measuring the radiation levels in the vicinity of or close to the Reactor Building;
- the duty shift engineer posts the appropriate radiological control signs at the entrance of the Reactor Building; and
- before entry to the refueling area is made, a careful study of conditions, radiation levels, etc., is performed.

15.4.1.2.1 System Operation

Normally operating plant instrumentation and controls are assumed to function, although credit is taken only for the isolation of the normal ventilation system. Operation of other plant reactor protection or engineered safety feature (ESF) systems is not expected.

15.4.1.3 Core and System Performance

15.4.1.3.1 Mathematical Model

The analytical methods and associated assumptions used to evaluate the radiological consequences of this accident are based on NUREG-1465 alternative source terms (AST) and the methodology in Regulatory Guide (RG) 1.183, to demonstrate compliance with the 10 CFR 50.67, SRP 15.0.1 and RG 1.183 total effected dose equivalent (TEDE) acceptance criteria.

15.4.1.3.2 Input Parameters and Initial Conditions

Regulatory Guide (RG) 1.183 provides assumptions acceptable to the NRC that may be used in evaluating the radiological consequences of a postulated fuel-handling accident resulting in damage to the fuel cladding and subsequent release of radioactive materials.

15.4.1.3.3 Number of Failed Fuel Rods

The number of fuel bundles that are assumed to be damaged as a result of the fuel handling accident is provided in Table 15.4-2. It is assumed that all of the fuel rods have damaged fuel cladding in each fuel bundle that is damaged as a result of the fuel handling accident.

15.4.1.4 Radiological Consequences

Radiological analysis is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10 CFR 50.34 guidelines.

The fission product inventory in the fuel rods that are assumed to be damaged is based on the days of continuous operation at full power. Due to plant cool down and disassembly operations, there is a time delay following initiation of reactor shutdown before fuel movement operations can be initiated. The analysis is based on Regulatory Guide 1.183. Specific values or parameters used in the evaluation are presented in Table 15.4-2.

15.4.1.4.1 Fission Product Transport to the Environment

The emergency protection guides require that under FHA conditions the HVAC system be shut down and the fuel-handling area of the Reactor Building or Fuel Building isolated. Following isolation, the operator determines the extent of contamination and time for resuming operation of the HVAC. However, for the purposes of conservative estimation, no credit is taken for isolation or use of the HVAC System, and the gases are directly released to the environment at the rate identified in Table 15.4-3. The total activity released to the environment is presented in Table 15.4-3.

15.4.1.4.2 Assumptions to be Confirmed by the COL Applicant

The following are assumptions in the radiological analysis that require confirmation by the COL applicant:

- Normal intake flow rate into the Control Room is 500 cfm.
- Emergency recirculation flow rate into and out of the Control Room is 250 cfm.
- The atmospheric dispersion factor for the Control Building is 1.0E-03 s/m^3.
- The Control Building HVAC system control system lag time is 10 seconds.
- The Control Room Habitability Area volume is 92,700 ft^3.
- Fuel handling operations do not occur until 24 hours after shutdown.

15.4.1.5 Results

The results of this analysis are presented in Tables 15.4-4 for both offsite and control room dose evaluations and are within 10 CFR 50.67 and RG 1.183 regulatory guidelines.

15.4.2 Loss-of-Coolant Accident Containment Analysis

The containment performance analysis is provided within Section 6.2, and demonstrates that containment systems meet their design limits for all postulated design basis events.

15.4.3 Loss-of-Coolant Accident ECCS Performance Analysis

The emergency core cooling system (ECCS) performance analysis evaluates the full spectrum of pipe breaks, including the worst case of piping break inside containment. This analysis is provided within Section 6.3, and demonstrates compliance with the 10 CFR 50.46 ECCS acceptance criteria.

15.4.4 Loss-of-Coolant Accident Inside Containment Radiological Analysis

This event assumes a worst case of piping break inside containment. This event is in part based on the fact that the ECCS performance analysis demonstrates to what level that the 10 CFR 50.46 ECCS acceptance criteria are met, and that the containment analysis demonstrates that containment systems meet their design limits. The following discussion provides only information not presented in the subject sections. All other information is cross-referenced.

The postulated event represents the envelope evaluation for liquid or steam line failures inside containment.

The following analysis is based on NUREG-1465 alternative source terms (AST) and the methodology in Regulatory Guide (RG) 1.183, and demonstrates compliance with the 10 CFR 50.67, SRP 15.0.1 and RG 1.183 total effected dose equivalent (TEDE) acceptance criteria.

15.4.4.1 Identification of Causes

There are no realistic, identifiable events that would result in a pipe break inside the containment of the magnitude required to cause a LOCA coincident with a Safe Shutdown Earthquake (SSE). The subject piping is of high quality, designed to nuclear construction industry codes and standards, and for seismic and environmental conditions. However, because such an accident provides an upper limit estimate for the resultant effects for this category of pipe breaks, it is evaluated without the causes being identified.

15.4.4.2 Sequence of Events and Systems Operation

15.4.4.2.1 Sequence of Events

The sequence of events associated with this accident is presented in Section 6.3 for ECCS performance and Section 6.2 for barrier (containment) performance.

Following the pipe break and scram, the MSIVs close on the reactor water low level trip signal (Level 2). Some moments later, the reactor low water (Level 1) signal initiates the ADS and GDCS. The core remains covered throughout the accident and there is no fuel damage.

15.4.4.2.2 Identification of Operator Actions

Because automatic actuation and operation of the ECCS is a system design basis, no operator actions are required. However, the operator should perform the following:

- Verify that all rods have inserted;
- Monitor reactor water level and pressure;
- Verify ADS actuation on Level 1;
- Verify GDCS flow at low vessel pressure by observing check valve at open position and GDCS pool level decreasing; and
- Periodically monitor the oxygen concentration in the drywell and wetwell.

15.4.4.2.3 Systems Operations

For all design basis LOCA events described within Section 6.3, there is no core uncovery or heatup. Therefore, no fission products other than spiking terms associated with rapid depressurization occur. Multiple failures of safety-related systems are required to cause significant core damage. Nevertheless, a fission product release is assumed without regard to mechanistic causes to evaluate the ability of the design to mitigate potential fission product releases to the containment.

Accidents that could result in the release of radioactive fission products directly into the containment are the results of postulated reactor coolant pressure boundary (RCPB) pipe breaks. All pipe breaks, sizes and locations are presented in Sections 6.2 and 6.3, including the severance of main steam lines, emergency core cooling system lines, feedwater lines and other process system lines. The minimum required functions of any reactor and plant protection system are presented in Sections 6.2, 6.3, 7.3, 7.6 and 8.3.

15.4.4.3 Core and System Performance

15.4.4.3.1 Mathematical Model

The analytical methods and associated assumptions that are used in evaluating the consequences of this accident are considered to provide a conservative assessment of the consequences of this improbable event. The details of these calculations, their justification and bases for the models are developed to comply with RG 1.183.

15.4.4.3.2 Input Parameters and Initial Condition

Input parameters and initial conditions used for the analysis of this event are presented in Section 6.3.

15.4.4.3.3 Results

Results of this event are presented in detail within Section 6.3. The temperature and pressure transients resulting from this accident are insufficient to cause perforation of the fuel cladding. Therefore, no fuel damage results from this accident. Post-accident tracking instrumentation and control is assured. Continued long-term core cooling is demonstrated. Radiological effects are minimized and within limits. Continued operator control and surveillance is examined and provided.

15.4.4.4 Barrier Performance

The structural design basis for the containment is to maintain its integrity and experience normal stresses after the instantaneous rupture of any primary system piping within the structure, while also accommodating the dynamic effects of the pipe break and an SSE. Therefore, any postulated LOCA does not result in exceeding the containment design limit (see Subsection 3.8.2.3, Section 3.6 and Section 6.2 for details and results of the analyses).

15.4.4.5 Radiological Consequences

One analysis is provided for the evaluation of the radiological consequences of a design basis loss-of-coolant accident (LOCA), for both the offsite dose evaluations and the control room dose evaluations. The analysis is based upon a process flow diagram shown in Figure 15.4-1 and accident parameters specified in Table 15.4-5. The analysis is based upon assumptions provided in Regulatory Guide 1.183 except where noted.

15.4.4.5.1 Radionuclide Releases and Pathways

Two specific pathways are analyzed in releasing radionuclides to the environment. The first pathway is leakage to the reactor building via penetrations. This leakage pathway is assumed to be no greater than an equivalent release of 0.5% volume per day from the containment per plant Technical Specifications**.** The reactor building leaks to the environment at a rate specified in Table 15.4-5.

The second leakage pathway is MSIV leakage to the turbine building condenser. This pathway is discussed separately below.

15.4.4.5.2 Passive Containment Cooling System (PCCS)

The PCCS is used to condense steam and control pressure in the event of a LOCA. The PCCS effectively scrubs the containment atmosphere by removing fission products from the containment atmosphere from condensation of the steam. The PCCS decontamination factor assumed for removal of radioiodines is provided in Table 15.4-5.

Prior studies of expected chemical species post-LOCA in a BWR (Reference 15.4-3) show that primarily due to CsOH, the pH of BWR pools falls into the range of 9.5 to 10.5 at the onset of radionuclide release. Over a period of days then, the pH is expected to slowly fall from radiolysis of the nitrogen atmosphere and production of nitric acid. To prevent the pH from falling into ranges below 7.0, water buffering agents can be injected into the pool through any of the following systems:

- RWCU/SDC
- FAPCS
- SLC
- Feedwater
- CRD Makeup

Because of the above redundancy, it is expected that buffering agents would be added under degraded core conditions, and re-evolution of iodine species would not occur because of lower pH in the water.

15.4.4.5.3 Main Steamline Modeling

The second potential release pathway is via the main steamline through leakage in the main steamline isolation valves. It is assumed that a pathway exits which permits the containment atmosphere, or in the non-break case, pressure vessel air space direct access to the main steamlines and that the main steamline isolation valves leak at the Technical Specification limit. Furthermore, it is assumed that the most critical main steamline isolation valve fails in the open position. Therefore, the total leakage through the steamlines is equal to the Technical Specification limit for the plant.

The main steamlines are classified (see Table 3.2-1) as Seismic Category I from the pressure vessel interface to the outboard seismic restraint outboard of the downstream MSIV, thereby providing a qualified safety-related mitigation system for fission product leakage, which, in this case, is limited by the leakage criteria specific in the technical specifications for the MSIVs. The primary purpose of this system is to stop any potential flow through the main steamlines. Downstream of the seismic restraint referred to above, the steamlines pass through the Reactor Building - Turbine Building interface into the Turbine Building steam tunnel. The Turbine Building steam tunnel is a heavily shielded reinforced concrete structure designed to shield workers from main steamline radiation shine. The steamlines and their associated branch lines outboard of the last Reactor Building seismic restraint are Quality Group B structures. In addition, these lines and structures are required to be dynamically analyzed to SSE conditions (Table 3.2-1) that determine the flexibility and structural capabilities of the lines under hypothetical SSE conditions.

The analysis of leakage from the containment through the main steamlines involves the determination of

- probable and alternate flow pathways,
- physical conditions in the pathways, and
- physical phenomena that affect the flow and concentration of radionuclides in the pathways.

The most probable pathway for radionuclide transport from the main steamlines is found to be from the outboard MSIVs into the drain lines coming off the outboard MSIV and then into the turbine building to the main condenser. A secondary path is found along the main steamlines into the turbine though flow through this pathway as described below is a minor fraction of the flow through the drain lines.

Consideration of the main steamlines and drain line complex downstream of the reactor building as a mitigative factor in the analysis of LOCA leakage is based upon the following determination:

- The main steamlines and drain lines are high quality lines inspected on a regular schedule.

- The main steamlines and drain lines are designed to meet SSE criteria and analyzed to dynamic loading criteria.
- The main steamlines and drain lines are enclosed in a shielded corridor that protects them from collateral damage in the event of an SSE. For those portions not enclosed in the steam tunnel complex, an as-built inspection is required to verify that no damage could be expected from other components and structures in a SSE.
- The main steamlines and drain lines are required under normal conditions to function to loads at temperature and pressure far exceeding the loads expected from an SSE. This capability inherent in the basic design of these components furnishes a level of toughness and flexibility to ensure their survival under SSE conditions. A large database of experience in the survival of these types of components under actual earthquake conditions proves this contention (Reference 15.4-4). In the case of the ESBWR, further margin for survival can be expected, because the ESBWR lines are designed through dynamic analysis to survive such events, whereas in the case of the actual experience database, the lines shown to survive were designed to lesser standards to meet only normally expected loads.

Based upon the facts above, the main steamlines and drain lines are used as mitigative components in the analysis of leakage from the MSIVs.

The analysis of leakage from the MSIVs follows the procedures and conditions specified in Reference 15.4-4.

15.4.4.5.4 Condenser Modeling

The condenser is modeled as detailed in Reference 15.4-4 with specific values used given in Table 15.4-5. The volume is modeled primarily as a stagnant volume assuming the shutdown of all active components. The condenser is used as a mitigative volume based upon the determination that such components, designed to standard engineering practice, are sufficiently strong to withstand SSE conditions due wholly to their design (Reference 15.4-4). The only requirement in the design of the condenser is that it be bolted to the building basemat to prevent walking during an earthquake. The turbine is bolted to the turbine floor with the condenser bolted to the turbine and suspended off the turbine. The requirement on these components for purposes of mitigation is only that they survive as a volume and not that they provide functionality or leak tightness following an earthquake.

Releases from the condenser/turbine building pathway are assumed via diffuse sources in the Turbine Building. The two major points of release in the Turbine Building are expected to be the truck doors at the far end of the Turbine Building and the Turbine Building vent panels located midway on the Turbine Building on the side away from the Reactor Building. Releases are assumed to be ground level releases.

15.4.4.5.5 Control Room

The control room is physically integrated into the Control Building and is located below grade adjacent to the Reactor, Service, and Turbine Buildings. During a LOCA, exposure to the operators consists of contributions from airborne fission products entrained into the control room ventilation system.

Exposure to the operators from airborne contamination consists almost entirely of radionuclides entrained into the control room environment via the HVAC from the atmosphere. The control room is designed to operate under minimal power and HVAC conditions with air flow into the control room and positive pressure maintained by a redundant bottled air supply system. The system can maintain the control room under positive pressure minimal inleakage conditions for 72 hours prior to requiring recharging. Recharging can be done at the site or bottled air can be trucked into and installed in the air supply bays in the Emergency Breathing Air System Building. The bottled air system is described in Subsection 6.4.3.

In addition to the bottled air system, the HVAC can also operate under positive flow with intake air routed through a single train charcoal adsorber system. For purposes of conservative calculations, it is assumed that for the first 72 hours the control room is maintained under bottled air, transferring over to the HVAC charcoal train after 72 hours. This proves to be the most conservative assumptions, because if the bottled air supply is replenished each 72 hours, the primary mode of introducing radionuclides into the control room, the HVAC system, is not turned on. In the event that the HVAC fails after the initial 72 hours, the bottled air system would be resumed.

Control room dose is based upon fission product releases modeled as described above and the values presented in Table 15.4-8. Operator exposure is also based upon those conditions given in Table 15.4-8. The occupancy factors shown below are derived from RG 1.183.

Time	Occupancy Factor
0-1 day	1.0
1-4 days	0.6
> 4 days	0.4

15.4.4.5.6 Meteorology and Site Assumptions

Offsite Meteorology - This DCD uses a generic U.S. site that does not specifically identify meteorological parameters adequate to define dispersion conditions for accident evaluation. Therefore, a set of dispersion parameters (χ/Q's) were selected to simulate a U.S. site, which are given in Table 15.4-9 for the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ).

Control Room Meteorology - No specific acceptable method exists to calculate the meteorology for standard plant application for control room dose analysis. The control room assumed dispersion factor (χ/Q) is provided in Table 15.4-9.

15.4.4.5.7 Assumptions to be Confirmed by the COL Applicant

The following are assumptions in the radiological analysis that require confirmation by the COL applicant:

- A decontamination factor of 10 (efficiency = 90%) is used for the pathway from the Drywell to the PCCS volume.
- The primary containment leakage rate is reduced to 0 volume % per day after 72 hours.

- The Reactor Building leakage rate is assumed to be 100 volume % per day.
- Because the Control Room is sealed, there is no unfiltered inleakage term.
- The emergency breathing air supply (EBAS) isused in the Control Room for the first 72 hours following the LOCA. The Control Room HVAC then runs in emergency filtration mode for the remaining duration of the event.

15.4.4.6 Results

The results of this analysis are presented in Table 15.4-9 for both offsite and control room dose evaluations and are within 10 CFR 50.63 and RG 1.183 regulatory guidelines. Additional detail on the distribution of iodine is provided in Appendix 15B.

15.4.5 Main Steamline Break Accident Outside Containment

This event involves postulating a large steam line pipe break outside containment. It is assumed that the largest steam line instantaneously and circumferentially breaks at a location downstream of the outermost isolation valve. The plant is designed to immediately detect such an occurrence, initiate isolation of all main steamlines including the broken line and actuate the necessary protective features. This postulated event represents the envelope evaluation of steam line failures outside containment.

The Main Steamline Break Accident (MSLBA) containment response evaluation is provided in Section 6.2

The MSLB ECCS capability evaluation is provided in Section 6.3.

The MSLB radiological evaluation is as follows:

15.4.5.1 Identification of Causes

A MSLBA is postulated without the cause being identified. These lines are designed to high quality engineering codes and standards, and to seismic and environmental requirements. However, for the purpose of evaluating the consequences of a postulated large steam line rupture, the failure of a main steam line is assumed to occur.

15.4.5.2 Sequence of Events and Systems Operation

15.4.5.2.1 Sequence of Events

Accidents that result in the release of radioactive materials directly outside the containment are the result of postulated breaches in the reactor coolant pressure boundary or the steam power conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting event for breaks outside the containment is a complete severance of one of the main steamlines. The sequence of events and approximate time required to reach the event is given in Table 15.4-10.

Following isolation of the main steam supply system (i.e., MSIV closure), the ADS initiates automatically on low water level (Level 1). Once the reactor system has depressurized, the GDCS automatically begins reflooding the reactor vessel. The core remains covered throughout the accident, and there is no fuel damage.

15.4.5.2.2 Systems Operation

A postulated guillotine break of one of the main steam lines outside the containment results in mass loss from both ends of the break. The flow from the upstream side is initially limited by the flow restrictor within the reactor vessel steam outlet nozzle. Flow from the downstream side is initially limited by the flow restrictor within the reactor vessel steam outlet nozzle for the three unbroken lines. Subsequent closure of the MSIVs further limits the flow when the valve area becomes less than the limiter area and finally terminates the mass loss when the full closure is reached.

A discussion of plant and RPS action and ESF action is presented in Sections 6.3, 7.3 and 7.6.

15.4.5.2.3 The Effect of Single Failures and Operator Errors

The steamline break outside the containment is a special case of the general LOCA break spectrum considered in detail in Section 6.3. The general single-failure analysis for LOCAs is presented in Subsection 6.3.3. For the steamline break outside the containment, the worst single failure does not result in core uncovery (see Section 6.3 for analysis details).

15.4.5.3 Core and System Performance

Quantitative results (including mathematical models, input parameters, and consideration of uncertainties) for this event are presented in Section 6.3. The temperature and pressure transient results from this accident are not sufficient to cause fuel damage.

15.4.5.3.1 Input Parameters and Initial Conditions

Input parameters and initial conditions used for the ECCS performance analysis of this event are presented in Table 6.3-1.

15.4.5.3.2 Results

There is no fuel damage as a result of this accident. Refer to Section 6.3 for ECCS analysis.

15.4.5.4 Barrier Performance

Because this break occurs outside the containment, barrier performance within the containment envelope is not applicable. Details of the results of this event can be found in Subsection 6.2.3.

Initially, only steam issues from the broken end of the steam line. The flow in each line is limited by critical flow at the limiter for each line. Rapid depressurization of the RPV causes the water level to rise, resulting in a steam-water mixture flowing from the break until the valves are closed. The total integrated mass leaving the RPV through the steamline break is provided in Table 15.4-11.

15.4.5.5 Radiological Consequences

The radiological analysis for this accident is based on conservative assumptions considered to be acceptable to the NRC for the purposes of determining adequacy of the plant design to meet 10 CFR 50.67 guidelines. This analysis is referred to as the "design basis analysis."

15.4.5.5.1 Design Basis Analysis

Specific values of parameters used in the evaluation are presented in Table 15.4-11.

General Compliance or Alternate Approach Statement (RG 1.183): This guide provides assumptions acceptable to the NRC that may be utilized in evaluating the radiological consequences of a MSLBA for a BWR.

Some of the models and conditions that are prescribed are inconsistent with actual physical phenomena. The effect of the conservative bias that is introduced is generally limited to plant design choices not within the scope of the ESBWR Standard Plant design. The resultant dose is within regulatory limits.

Fission Product Release from Fuel: There is no fuel damage as a result of this accident. The only activity available for release from the break is that which is present in the reactor coolant and steamlines prior to the break.

Fission Product Transport to the Environment: The transport pathway is a direct unfiltered release to the environment with an air exchange rate in the steam tunnel of 1.2E+06 exchanges per day. Assuming that all of the activity in the steam becomes airborne, the release of activity to the environment is presented in Table 15.4-12.

Assumptions to be Confirmed by the COL Applicant

The following are assumptions in the radiological analysis that require confirmation by the COL applicant:

- Normal intake flow rate into the Control Room is 500 cfm.
- Emergency recirculation flow rate into and out of the Control Room is 250 cfm.
- The atmospheric dispersion factor for the Control Building is 1.0E-03 s/m^3.
- The Control Building HVAC system control system lag time is 10 seconds.
- The Control Room Habitability Area volume is 92,700 ft^3.
- The estimated volume of the steam tunnel is 1.21E+12 mm^3.

15.4.5.5.2 Results

The calculated exposures for the design basis analysis are presented in Table 15.4-13 and are less than the guidelines of RG 1.183 and 10 CFR 50.67.

15.4.6 Control Rod Drop Accident

15.4.6.1 Features of the Fine Motion Control Rod Drives

As presented in Subsection 4.6.1, the fine motion control rod drive (FMCRD) has several new features that are unique compared with locking piston control rod drives.

In each FMCRD, there are dual Class 1E separation-detection devices that detect the separation of the control rod from the FMCRD if the control rod is stuck and separated from the ballnut of the FMCRD. The control rods are normally inserted into the core and withdrawn with the hollow piston, which is connected with the control rod, resting on the ballnut. The separation-detection device is used at all times to ascertain that the hollow piston and control rod are resting on the ballnut of the FMCRD. The separation-detection devices sense motion of a spring loaded support for the ball screw and in turn the hollow piston and the control rod. Separation of either the control rod from the hollow piston or the hollow piston from the ballnut is detected immediately. When separation has been detected, the interlocks preventing rod withdrawal operate to prevent further control rod withdrawal. Also, an alarm signal would be initiated in the control room to warn the operator.

There is also the unique highly reliable bayonet type coupling between the control rod blade and the FMCRD. With this coupling, the connection between the blade and the drive cannot be separated unless they are rotated 45 degrees. This rotation is not possible during reactor operation. There are procedural coupling checks to assure proper coupling. There is also the automated overtravel check in the RC&IS logic during automated operation. Finally, there is the latch mechanism on the hollow piston part of the drive. If the hollow piston is separated from the ballnut and rest of the drive due to stuck rod, the latch limits any subsequent rod drop to a short distance. More detailed descriptions of the FMCRD system are presented in Subsection 4.6.1. Failure modes of the FMCRD are discussed in Appendix 15A.

15.4.6.2 Identification of Causes

For the rod drop accident to occur, it is necessary for the following highly unlikely events to occur:

- the reactor is at < 5% power;
- there are failures of both Class 1E separation-detection devices or a failure of the rod block interlock;
- there is a failure of the latch mechanism;
- there is a failure that causes the rod to become stuck in the inserted position;
- the control rod drive is withdrawn without the operators noticing that the control rod withdrawal did not result in a neutron flux increase; and
- the stuck rod becomes unstuck and drops out of the core.

Alternatively, separation of the blade from the hollow piston would require either that control rod was installed without coupling and the coupling checks failed, or there is structural failure of this coupling. Under circumstances of acoupling failure, the rod drop accident can only occur with the simultaneous failure of both separation-detection devices (or the failure of the rod block interlock), together with the occurrence of a stuck rod on the same FMCRD.

In either case, because of the low probability of such simultaneous occurrence of these multiple independent events, there is no basis to postulate this event to occur.

15.4.6.3 Sequence of Events and System Operation

15.4.6.3.1 Sequence of Events

The bayonet coupling and procedural coupling checks preclude the uncoupling of the control rod from the hollow piston of the FMCRD. If the control rod is stuck, the separation-detection devices would detect the separation of the control rod and hollow piston from the ballnut of the FMCRD and rod block interlock would prevent further rod withdrawal. The operator would be alarmed of this separation.

Therefore, there is no technical basis for the control rod drop event to occur.

15.4.6.3.2 Identification of Operator Actions

No operator actions are required to preclude this event. However, the operator would be notified by the separation-detection alarm if separation is detected.

15.4.6.4 Core and System Performance

The performance of the separation-detection devices and the rod block interlocks virtually preclude the cause of a rod drop accident.

15.4.6.5 Barrier Performance

An evaluation of the barrier performance is not made for this accident, because there is no circumstance for which this event could occur.

15.4.6.6 Radiological Consequences

The radiological analysis is not required.

15.4.7 Feedwater Line Break Outside Containment

The feedwater line break containment response evaluation is provided in Section 6.2

The feedwater line break ECCS capability evaluation is provided in Section 6.3.

The feedwater line break radiological evaluation is as follows:

The postulated break of the feedwater line, representing the largest liquid line outside the containment, provides the envelope evaluation for this type of break. The break is assumed to be instantaneous, circumferential and downstream of the outermost isolation valve.

A more limiting event from a core performance evaluation standpoint (Feedwater Line Break Inside Containment) has been quantitatively analyzed and is presented in Section 6.3. Therefore, the following discussion provides only new information not presented in Section 6.3. All other information is cross-referenced to appropriate Chapter 6 subsections.

15.4.7.1 Identification of Causes

A feedwater line break is assumed without the cause being identified. The subject piping is designed to high quality, to engineering codes and standards, and to seismic environmental requirements.

15.4.7.2 Sequence of Events and System Operation

15.4.7.2.1 Identification of Operator Actions

Because automatic actuation and operation of the ECCS is a system design basis, no operator action is required.

However, the operator should perform the following (shown for informational purposes only):

- Determine that a line break has occurred and evacuate the area of the Turbine Building.
- Ensure that the reactor is shut down and that the ICS and the CRD systems are operating normally or, if failed, that the ADS and GDCS are operating.
- Implement site radiation incident procedures.
- Shut down the feedwater system and de-energize any electrical equipment that may be damaged by water from the feedwater system in the turbine building.
- Continue to monitor reactor water level and the performance of the ECCS while the radiation incident procedure is being implemented and begin normal reactor cooldown measures.
- Initiate the FAPCS in the suppression pool cooling mode (if necessary) and RWCU/SDC in the shutdown cooling mode.

These actions occur over an elapsed time of 3-4 hours.

15.4.7.2.2 Systems Operation

It is assumed that the normally operating plant instrument and controls are functioning. Credit is taken for the actuation of the ECCS. The Reactor Protection System, SRVs, ECCS, and Control Rod Drive system are assumed to function properly to ensure a safe shutdown.

The ESF systems, including the ADS and GDCS, are assumed to operate normally.

15.4.7.2.3 The Effect of Single Failures and Operator Errors

The feedwater line outside the containment is a special case of the general LOCA break spectrum presented in detail within Section 6.3. The general single-failure analysis for LOCAs is presented in detail in Subsection 6.3.3. For the feedwater line break outside the containment, the worst single failure does not result in core uncovery, and there would be no fuel damage.

15.4.7.3 Core and System Performance

15.4.7.3.1 Qualitative Summary

The accident evaluation qualitatively considered in this subsection is considered to be a conservative and envelope assessment of the consequences of the postulated failure (i.e., severance) of one the feedwater piping lines external to the containment.

15.4.7.3.2 Qualitative Results

The feedwater line break outside containment is less limiting, from a core performance evaluation standpoint, than the main steamline break outside the containment analysis presented in Subsection 15.4.5 and the LOCA inside the containment analysis presented in and 15.4.4.

The break is isolated by closure of the feedwater check valves. The main steamlines are isolated on water level 2, and the ADS and the GDCS together restore the reactor water level to the normal elevation. The fuel is covered throughout the transient and there is no pressure or temperature transient sufficient to cause fuel damage.

15.4.7.3.3 Consideration of Uncertainties

This event is conservatively analyzed and uncertainties were adequately considered (see Section 6.3).

15.4.7.4 Barrier Performance

Accidents that result in the release of radioactive materials outside the containment are the result of postulated breaches in piping connected to the reactor coolant pressure boundary or the steam power-conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting fault event for breaks outside the containment is a complete severance of one of the main steamlines as presented in Subsection 15.4.5. The feedwater system piping break is less severe than the main steamline break. Results of analysis of this event can be found in Section 6.3.

15.4.7.5 Radiological Consequences

15.4.7.5.1 Analysis

The NRC provides no specific regulatory guidelines for the evaluation of this accident; therefore, the analysis presented is based upon conservative assumptions considered acceptable to the NRC. However, for consistency, the RG 1.183 guideline exposure acceptance criteria for the MSLBA are used for the Feedwater Line Break accident.

The analysis is based on a conservative assessment of this accident. The specific models, assumptions and the program used for computer evaluation are presented in Reference 15.4-1. Specific values of parameters used in the evaluation are presented in Table 15.4-14.

15.4.7.5.2 Fission Product Release

There is no fuel damage as a consequence of this accident. In addition, an insignificant quantity of activity (compared to that existing in the main condenser hotwell prior to the occurrence of the break) is released from the contained piping system prior to isolation closure.

The iodine concentration assumed is that of the maximum equilibrium reactor water concentration used for the MSLBA, subject to a 2% carryover of iodine in the water to steam condensate. Noble gas activity in the condensate is negligible, because the air ejectors remove all noble gases from the condenser.

15.4.7.5.3 Fission Product Transport to the Environment

The transport pathway consists of liquid release from the break, carryover to the turbine building atmosphere due to flashing and partitioning and unfiltered release to the environment through the turbine building ventilation system.

Taking no credit for holdup, decay or plate-out during transport through the turbine building, the release of activity to the environment is presented in Table 15.4-15.

15.4.7.5.4 Assumptions to be Confirmed by the COL Applicant

The following are assumptions in the radiological analysis that require confirmation:

- The main condenser is sized for at least 2 minutes worth of main steam flow.

- The demineralizer efficiency is at least 99% (all coolant that is released during the accident is filtered through the demineralizer).

15.4.7.5.5 Results

The calculated exposures for the analysis are presented in Table 15.4-16, and are less than the regulatory guideline exposures.

15.4.8 Failure of Small Line Carrying Primary Coolant Outside Containment

This event postulates a small steam or liquid line pipe break inside or outside the containment, but within a controlled release structure. To bound the event, it is assumed that a small instrument line, instantaneously and circumferentially, breaks at a location where it may not be able to be isolated and where detection is not automatic or apparent. This event is less limiting from a core performance evaluation standpoint than the postulated events presented in Subsections 15.4.5 (Main Steamline Break Accident Outside Containment), 15.4.4 (Loss-of-Coolant Accident Inside Containment Radiological Analysis), and 15.4.7 (Feedwater Line Break Outside Containment).

This postulated event represents the envelope evaluation for small line failure inside and outside the containment relative to sensitivity for detection.

15.4.8.1 Identification of Causes

There is no identified specific event or circumstance that results in the failure of an instrument line. These lines are designed to high quality, engineering standards, seismic and environmental requirements. They also are equipped with either excess flow check valves or isolation valves. However, for the purpose of evaluating the consequences of a small line rupture, the rupture of an instrument line is assumed to occur along with a failure to isolate the break.

A circumferential rupture of an instrument line that is connected to the primary coolant system is postulated to occur outside the drywell, but inside the Reactor Building. The associated effects from a rupture in the drywell would not be as significant as those from the failure in the Reactor Building.

15.4.8.2 Sequence of Events and Systems Operations

15.4.8.2.1 Sequence of Events

The leak may result in noticeable increases in radiation, temperature, humidity, or audible noise levels in the Reactor Building or abnormal indications of actuations caused by the affected instrument.

Termination of the analyzed event is dependent on operator action. The action is initiated with the discovery of the unisolatable leak. The action consists of the orderly shutdown and depressurization of the reactor.

15.4.8.2.2 Systems Operation

A presentation of plant, RPS, ESF and other safety-related action are given in Sections 6.3, 7.3 and 7.6.

15.4.8.2.3 The Effect of Single Failures and Operator Errors

There is no single failure or operator error that significantly affects the system response to this event. Single failures in other instrument channels could lead to actuation of ESF actions such as reactor scram or MSIV closure under the assumption that the line break trips one division and the single failure trips another division to produce a 2-out-of-4 trip condition.

15.4.8.3 Core and System Performance

Instrument line breaks, because of their small size, are substantially less limiting from a core and systems performance standpoint than the events examined in Subsections 15.4.4, 15.4.5 and 15.4.7. Consequently, instrument line breaks are considered to be bounded specifically by the MSLBA (Subsection 15.4.5). Details of this calculation, including those pertinent to core and system performance, are presented in Subsection 15.4.5.3.

15.4.8.3.1 Input Parameters and Initial Conditions

Input parameters and initial conditions used for the analysis of this event are presented in Table 15.4-17.

15.4.8.3.2 Results

No fuel damage or core uncovery occurs as a result of this accident. Instrument line breaks are within the spectrum considered in ECCS performance calculations presented in Section 6.3.

15.4.8.4 Barrier Performance

The following assumptions and conditions are the basis for the mass loss during the release period of this event.

- The instrument line releases coolant into the Reactor Building for 30 minutes at normal operating temperature and pressure. Following this time period, the operator detects the event, scrams the reactor and initiates reactor depressurization.
- Reactor coolant is released to the Reactor Building, until the reactor is depressurized.
- The flow from the instrument line is limited by reactor pressure and a 6-mm (0.25-inch) diameter flow restricting orifice inside the drywell. The Moody critical blowdown model is applicable, and the flow is critical at the orifice (Reference 15.4-6).

15.4.8.5 Radiological Analysis

15.4.8.5.1 General

The radiological analysis is based upon conservative assumptions considered acceptable to the NRC. Though the Standard Review Plan does not provide detailed guidance, the assumptions found in Table 15.4-17 assume that all of the iodine available in the flashed water is transported via the reactor building HVAC system to the environment without prior treatment. Other isotopes in the water contribute only negligibly to the iodine dose.

15.4.8.5.2 Fission Product Release

The iodine activity in the coolant is assumed to be at the maximum equilibrium Technical Specification limit (see MSLBA in Subsection 15.4.5.5) for continuous operation. Based on data in Table 15.4-17, the amount of iodine released to the Reactor Building atmosphere and to the environment is presented in Table 15.4-18.

15.4.8.5.3 Results

The calculated exposures for the analysis are presented in Table 15.4-19, and are less than the regulatory guideline exposures.

15.4.9 RWCU/SDC System Line Failure Outside Containment

15.4.9.1 Identification of Causes

To evaluate liquid process line pipe breaks outside containment, the failure of a cleanup water line is assumed to evaluate the response of the plant design to this postulated event. The postulated break of the cleanup water line, representing the most significant liquid line outside containment, provides the envelope evaluation for this type of break. The break is assumed to be instantaneous, circumferential and downstream of the outermost isolation valve.

15.4.9.2 Sequence of Events and Systems Operation

15.4.9.2.1 Sequence of Events

The sequence of events is presented in Table 15.4-20.

15.4.9.2.2 Identification of Operator Actions

Because automatic actuation and operation of the ECCS is a system design basis, no operator actions are required. However, the operator should perform the following (shown for informational purposes only):

- determine that a line break has occurred
- ensure that if vessel water level is below level 3 that reactor has scrammed,
- monitor vessel water level and ensure actuation of ECCS as needed, and
- implement site radiation incident procedures.

These actions occur over an elapsed time of 3–4 hours.

15.4.9.2.3 Systems Operation

It is assumed that the normally operating plant instrument and controls are functioning. Credit is taken for the actuation of the ECCS. The Reactor Protection System, SRVs, ECCS, and Control Rod Drive system are assumed to function properly to ensure a safe shutdown.

The ESF systems, including the ADS and GDCS, are assumed to operate normally.

15.4.9.3 Core and System Performance

The fuel is covered throughout the transient and there are no pressure or temperature transients sufficient to cause fuel damage.

15.4.9.4 Barrier Performance

Accidents that result in the release of radioactive materials outside the containment are the result of postulated breaches in piping connected to the RCPB or the steam power-conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting fault event for breaks outside the containment is a complete severance of one of the main steamlines as presented in Subsection 15.4.5. The cleanup water system piping break is less severe than the main steamline break.

15.4.9.5 Radiological Consequences

15.4.9.5.1 General

The NRC provides no specific regulatory guidelines for the evaluation of this accident; therefore, the analysis presented is based upon conservative assumptions considered acceptable to the NRC.

Specific values of parameters used in the evaluation are presented in Table 15.4-21.

15.4.9.5.2 Fission Product Release

There is no fuel damage as a consequence of this accident.

At the initiation of this accident it is assumed that the total non-filtered inventory in both the regenerative and non-regenerative heat exchangers is released through the break. Inventory in the demineralizer is prevented from being released by back flow check valves from exiting that component. A break on the downstream side of the demineralizer would be bounded due to the demineralizer action compared to a break on the upstream side of the demineralizer.

Isolation of the line is conservatively analyzed based upon actuation of the flow differential pressure instrumentation. This instrumentation has a built in 45 second time delay so that for the initial 45 seconds of the accident full flow exists through the line. After the initial 45 second flow, motor operated isolation valves close over a period of 30 seconds. During this period of 75 seconds, flow of reactor water is assumed at the maximum equilibrium reactor water concentration, with flashing to steam at reactor temperature and pressure. In addition, iodine spiking is assumed. Noble gas activity in the reactor coolant is negligible and is therefore ignored in this analysis.

15.4.9.5.3 Fission Product Transport to the Environment

It is conservatively assumed that the release to the environment is instantaneous, with no iodine plateout. Fission product releases to the environment are presented in Table 15.4-22.

15.4.9.5.4 Assumptions to be Confirmed by the COL Applicant

The following are assumptions in the radiological analysis that require confirmation by the COL applicant:

- The initial flow rate for the RWCU/SDC system line break accident is no greater than 193,000 lbm/hr.
- The water masses in the RHX and NRHX are no greater than 7500 lbm and 2900 lbm, respectively. Applying the assumed flashing fractions (0.279 for RHX and 0.074 for NRHX) to the masses released from the heat exchangers yields a total of 2315 lbm coolant flashed to steam.
- The mass of coolant released from the RHX and NRHX that is flashed to steam is no greater than 2315 lbm (flashing fractions of 0.279 for RHX and 0.074 for NRHX are used to determine the total mass of coolant flashed to steam).

15.4.9.5.5 Results

The calculated exposures for the analysis are presented in Table 15.4-23 and are less than the regulatory guideline exposures.

15.4.10 Spent Fuel Cask Drop Accident

15.4.10.1 Identification of Causes

Due to the redundant nature of the cask handling crane, the cask drop accident is not believed to be a credible accident. However, the accident is assumed to occur as a consequence of an unspecified failure of the cask lifting mechanism, thereby allowing the cask to fall from the level of the refueling floor to ground level through the refueling floor maintenance hatch.

15.4.10.2 Radiological Analysis

The largest size of ESBWR fuel cask is conservatively assumed to be dropped from the refueling floor level to ground level on transport from the decontamination pit out of the Reactor Building. It is conservatively assumed that all fuel rods inside the cask are damaged and the fission gases in the fuel rod gap space are released to the environment instantaneously. Table 15.4-24 provides the assumptions for this analysis and Table 15.4-25 provides the radiological releases. Table 15.4-26 demonstrates that the resulting exposures are within the regulatory guidelines.

15.4.11 COL Information

- FHA analysis assumptions (Subsection 15.4.1.4)
- LOCA analysis assumptions (Subsection 15.4.4.5.7)
- Main steamline break analysis assumptions (Subsection 15.4.5.5.1)
- Feedwater line break analysis assumptions (Subsection 15.4.7.5.4)
- RWCU/SDC line break analysis assumptions (Subsection 15.4.9.5.4)
- steam tunnel complex configuration (Subsection 15.4.4.5.3)

15.4.12 References

15.4-1 General Electric Co., “Radiological Accident Evaluation - The CONAC04A Code,” NEDO-32708, August 1997.

15.4-2 Electric Power Research Institute, “Advanced Light Water Reactor Utility Requirements Document,” Volume III.

15.4-3 General Electric Company, “Anticipated Chemical Behavior of Iodine under LOCA Conditions,” NEDO-25370, January 1981.

15.4-4 GE Nuclear Energy, “BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems,” NEDC-31858P (GE proprietary), February 1991.

15.4-5 General Electric Company, “Alternatives to Current Procedures Used to Estimate Concentrations in Building Wakes,” 21st DOE/NRC Nuclear Air Cleaning Conference, pgs 714-729.

15.4-6 General Electric Company, “Maximum Two-Phase Vessel Blowdown from Pipes,” ASME Paper Number 65-WA/HT-1, March 15, 1965.

Table 15.4-1

Fuel Handling Accident Sequence of Events

Sequence of Events	Elapsed Time
Channeled fuel bundle is being handled by a crane over reactor core. Crane motion changes from horizontal and the fuel grapple releases, dropping the bundle. The channeled bundle strikes unchanneled bundles in the rack.	0
Some rods in both the dropped and struck bundles fail, releasing radioactive gases to the pool water.	0
Gases pass from the water to the Reactor Building, fuel handling area.	0
The Reactor Building ventilation system high radiation alarm alerts plant personnel.	0+
Operator actions begin.	0+

Table 15.4-2

FHA Parameters

I. Data and Assumptions Used to Estimate Source Terms	
A. Power level, MWt	4590
B. Plenum Activity Radioactivity for I-131, %Radioactivity for Kr-85, % Radioactivity for other noble gases and halogens, %	 8 10 5
C. Radial peaking factor for damaged rods	1.5
D. Duration of accident, hr	2
E. No. bundles damaged	4
F. Minimum time after shutdown to accident, hr	24
G. Average fuel exposure, MWd/MT	35,000
II. Data and Assumptions Used to Estimate Activity Released	
A. Species fraction	
Organic iodine, %	0.15
Elemental iodine, %	4.85
Particulate iodine, %	95
Noble gas, %	100
B. Pool Retention decontamination factor	
Iodine (effective)	200
Noble gas	1
C. Reactor Building release rate, %/hr (350 %/hr is a rate of 3.5 air exchanges per hour)	350
III. Dispersion and Dose Data	
A. Meteorology	1.00E-03 s/m^3
B. Dose conversion assumptions	RG 1.183
C. Activity inventory/releases	Table 15.4-3
D. Dose evaluations	Table 15.4-4

Table 15.4-3

FHA Isotopic Release to Environment

Isotope	Activity (MBq)
I-131	1.8E+08
I-132	1.4E+08
I-133	1.1E+08
I-134	6.2E+00
I-135	1.9E+07
Kr-85M	1.6E+08
Kr-85	6.0E+08
Kr-87	2.7E+04
Kr-88	5.1E+07
Xe-133	4.7E+10
Xe-135	1.3E+10

Table 15.4-4

FHA Analysis Results

Accident Location, Exposure Location and Time Duration	**Maximum Calculated TEDE** (rem)	**10 CFR 50.67 Acceptance Criterion TEDE** (rem)
Within Containment:		
Exclusion Area Boundary (EAB) for a 2-hour duration	4.4	6.3
Outer boundary of Low Population Zone (LPZ) for a 2-hour duration	4.4	6.3
Control Room dose for the duration of the accident	1.0	5.0

Table 15.4-5

Loss-of-Coolant Accident Parameters

I. Data and Assumptions used to estimate source terms.	
A. Power Level, MWt	4590
B. Fraction of Core Inventory Released	RG 1.183
C. Iodine Chemical Species	
Elemental, %	4.85
Particulate, %	95
Organic, %	0.15
D. PCCS Decontamination Factors	
Noble Gas	1
Elemental iodine	10
Particulates	10
Organic iodines	1
II. Data and Assumptions used to estimate activity released	
A. Primary Containment Leakage	
Leak rate, %/day	0.5
MSIV leakage (total all lines), cfm	0.58
B. Reactor Building Leakage	
Leak rate, %/day	100
C. Condenser Data	
Free air volume, ft^3	2.20E+05
Fraction of volume involved, %	20
Iodine removal factors	
Particulate, %	99.5
Elemental, %	99.5
Organic, %	0
III. Control Room Parameters	
A. Control Room Volume, ft^3	9.27E+04
B. Recirculation Rates	

Table 15.4-5

Loss-of-Coolant Accident Parameters

0-72 hours	
Unfiltered inflow, ft^3/min	0
> 72 hours	
Filtered in leakage, ft^3/min	500
Recirculation rate, ft^3/min	250
Filter efficiency, %	99
Unfiltered inflow, ft^3/min	0
IV. Dispersion and Dose Data	
A. Meteorology	Table 15.4-9
B. Method of Dose Calculation	RG 1.183
C. Dose Conversion Assumptions	RG 1.183
D. Activity / Inventory Releases	Tables 15.4-6, 7, 8
E. Dose Evaluations	Table 15.4-9

Table 15.4-6

LOCA Compartment Inventories

Drywell Inventory (MBq)								
Isotope	**30 min**	**2 hr**	**4 hr**	**8 hr**	**12 hr**	**1 d**	**4 d**	**30 d**
Kr-85	2.63E+09	5.25E+10	5.24E+10	5.23E+10	5.23E+10	5.21E+10	5.11E+10	4.70E+10
Kr-85m	5.39E+10	8.53E+11	6.25E+11	3.36E+11	1.81E+11	2.81E+10	4.01E+05	0.00E+00
Kr-87	8.54E+10	7.52E+11	2.52E+11	2.85E+10	3.22E+09	4.62E+06	0.00E+00	0.00E+00
Kr-88	1.40E+11	1.94E+12	1.19E+12	4.48E+11	1.69E+11	8.98E+09	2.06E+02	0.00E+00
I-131	9.14E+10	1.90E+11	1.90E+09	1.84E+09	1.81E+09	1.73E+09	1.31E+09	1.29E+08
I-132	1.25E+11	2.58E+11	1.53E+09	4.49E+08	1.34E+08	3.60E+06	1.33E-03	0.00E+00
I-133	1.85E+11	3.68E+11	3.47E+09	2.98E+09	2.60E+09	1.74E+09	1.55E+08	1.33E-01
I-134	1.40E+11	8.94E+10	1.85E+08	7.69E+06	3.25E+05	2.45E+01	0.00E+00	0.00E+00
I-135	1.68E+11	2.99E+11	2.44E+09	1.58E+09	1.03E+09	2.93E+08	1.51E+05	0.00E+00
Xe-133	4.32E+11	8.59E+12	8.49E+12	8.29E+12	8.10E+12	7.55E+12	4.99E+12	1.48E+11
Xe-135	1.47E+11	3.00E+12	2.58E+12	1.90E+12	1.40E+12	5.58E+11	2.26E+09	0.00E+00

Reactor Building Inventory (MBq)								
Isotope	**30 min**	**2 hr**	**4 hr**	**8 hr**	**12 hr**	**1 d**	**4 d**	**30 d**
Kr-85	1.36E+05	8.56E+06	2.88E+07	6.46E+07	9.48E+07	1.60E+08	8.99E+07	4.57E-04
Kr-85m	2.80E+06	1.39E+08	3.44E+08	4.15E+08	3.28E+08	8.66E+07	7.06E+02	0.00E+00
Kr-87	4.43E+06	1.23E+08	1.39E+08	3.52E+07	5.83E+06	1.42E+04	0.00E+00	0.00E+00
Kr-88	7.28E+06	3.17E+08	6.55E+08	5.53E+08	3.06E+08	2.76E+07	3.62E-01	0.00E+00
I-131	6.32E+06	5.79E+07	6.20E+07	5.32E+07	4.58E+07	3.00E+07	3.25E+06	1.77E-06
I-132	8.28E+06	6.54E+07	4.65E+07	2.07E+07	1.23E+07	6.30E+06	1.65E+05	0.00E+00
I-133	1.28E+07	1.12E+08	1.13E+08	8.61E+07	6.58E+07	3.02E+07	3.85E+05	0.00E+00
I-134	9.69E+06	2.72E+07	6.04E+06	2.22E+05	8.22E+03	4.26E-01	0.00E+00	0.00E+00
I-135	1.16E+07	9.12E+07	7.97E+07	4.56E+07	2.62E+07	5.08E+06	3.76E+02	0.00E+00
Xe-133	2.24E+07	1.40E+09	4.67E+09	1.02E+10	1.47E+10	2.32E+10	8.78E+09	1.44E-03
Xe-135	7.65E+06	4.75E+08	1.42E+09	2.35E+09	2.55E+09	1.72E+09	3.99E+06	0.00E+00

Condenser Inventory (MBq)								
Isotope	**30 min**	**2 hr**	**4 hr**	**8 hr**	**12 hr**	**1 d**	**4 d**	**30 d**
Kr-85	3.19E+03	9.33E+05	8.15E+06	3.26E+07	6.01E+07	1.44E+08	6.14E+08	3.30E+09
Kr-85m	6.54E+04	1.52E+07	9.72E+07	2.09E+08	2.08E+08	7.77E+07	4.82E+03	0.00E+00
Kr-87	1.04E+05	1.34E+07	3.93E+07	1.77E+07	3.70E+06	1.28E+04	0.00E+00	0.00E+00
Kr-88	1.70E+05	3.46E+07	1.85E+08	2.79E+08	1.94E+08	2.48E+07	2.47E+00	0.00E+00
I-131	1.67E+05	9.02E+06	2.84E+07	3.97E+07	4.16E+07	4.23E+07	4.25E+07	1.05E+07
I-132	2.14E+05	9.32E+06	2.08E+07	1.56E+07	1.15E+07	9.76E+06	4.74E+06	9.41E+03
I-133	3.38E+05	1.75E+07	5.17E+07	6.43E+07	5.98E+07	4.26E+07	5.02E+06	1.08E-02

Table 15.4-6

LOCA Compartment Inventories

I-134	2.56E+05	4.25E+06	2.76E+06	1.66E+05	7.47E+03	6.01E-01	0.00E+00	0.00E+00
I-135	3.06E+05	1.42E+07	3.65E+07	3.41E+07	2.38E+07	7.18E+06	4.90E+03	0.00E+00
Xe-133	5.25E+05	1.53E+08	1.32E+09	5.16E+09	9.32E+09	2.09E+10	6.00E+10	1.04E+10
Xe-135	1.79E+05	5.10E+07	4.00E+08	1.19E+09	1.62E+09	1.55E+09	2.73E+07	0.00E+00

Table 15.4-7

LOCA Integrated Environment

I-131 Inventory

Time (hr)	I-131 Inventory (MBq)
0	1.13E-04
0.5	5.08E+04
2	1.97E+06
4	7.34E+06
8	1.72E+07
12	2.56E+07
16	3.29E+07
24	4.49E+07
30	5.20E+07
36	5.79E+07
48	6.73E+07
60	7.45E+07
72	8.04E+07
84	8.47E+07
96	8.74E+07
288	9.85E+07
480	1.04E+08
720	1.08E+08

Table 15.4-8

LOCA Control Room Inventories

	Inventory (MBq)				
Isotope	**3.5 d**	**4 d**	**12 d**	**20 d**	**30 d**
Kr-85	5.41E+03	3.62E+03	1.35E+03	2.03E+03	2.66E+03
Kr-85m	2.72E-01	2.84E-02	0.00E+00	0.00E+00	0.00E+00
Kr-88	4.08E-04	1.46E-05	0.00E+00	0.00E+00	0.00E+00
I-131	1.47E+00	9.53E-01	1.75E-01	1.11E-01	5.64E-02
I-132	9.61E-02	6.26E-02	3.77E-03	5.55E-04	5.07E-05
I-133	2.48E-01	1.13E-01	6.87E-05	1.45E-07	0.00E+00
I-135	5.72E-04	1.10E-04	0.00E+00	0.00E+00	0.00E+00
Xe-133	5.64E+05	3.53E+05	4.60E+04	2.40E+04	8.41E+03
Xe-135	5.99E+02	1.60E+02	2.63E-05	0.00E+00	0.00E+00

Table 15.4-9

LOCA Inside Containment Analysis Results

Exposure Location	**Meteorology (s/m^3)**	**Maximum Calculated TEDE** (rem)	**10 CFR 50.67 Acceptance Criterion TEDE** (rem)
Exclusion Area Boundary (EAB)	1.00E-03	5.0	25
Outer boundary of Low Population Zone (LPZ)	1.35E-04 (0-8 h) 1.00E-04 (8-24 h) 5.40E-05 (1-4 d) 2.20E-05 (4-30 d)	5.6	25
Control Room	1.00E-03	0.2	5

Table 15.4-10

Sequence of Events for Main Steamline Break Accident (MSLBA) Outside Containment

Time (sec)	**Event**
0	Guillotine break of one main steam line outside containment.
0.5	High steamline flow signal initiates closure of MSIVs
< 1.0	Reactor begins scram.
< 2.0	Partial closure (15%) of MSIV initiates isolation condensers.
< 5	MSIVs fully closed.
10	Reactor low water Level 2 is reached. Isolation condensers receive second initiation signal.
32	Isolation condensers in full operation. Water level stabilized.
435	SRVs open on high vessel pressure (if isolation condensers are not available). The SRVs open and close to maintain vessel absolute pressure.
3540	Reactor low water Level 1 is reached (if isolation condensers are not available). ADS timer initiated.
3550	ADS timer timed out. ADS actuation sequence initiated. GDCS timer initiated.
3700	GDCS timer timed out. GDCS injection valves open.
3880	Vessel pressure decreases below shutoff head of GDCS. GDCS reflooding flow into the vessel begins.
* The core remains covered throughout the transient and no core heatup occurs.	

Table 15.4-11

MSLBA Parameters

1.	Data and assumptions used to estimate source terms	
	A. Fuel Damage	none
	B. Reactor Coolant Activity: Pre-incident Spike Equilibrium Iodine Activity	 4.0 µCi/g DE I-131 0.2 µCi/g DE I-131
	C. Steam Mass Released, kg (lbm)	5,646 (12,436)
	D. Water Mass Released, kg (lbm)	98,793 (217,606)
2.	Data and assumptions used to estimate activity released	
	A. Isolation valve closure time, sec	5
	B. Maximum release time, hr	Instantaneous
3.	Dispersion Data	
	A. Meteorology	1.00E-03 s/m^3
	B. Method of Dose Calculation	RG 1.183
	C. Dose Conversion Assumptions	RG 1.183
	D. Activity Inventory and Releases	Tables 15.4-12
	E. Dose Evaluations	Table 15.4-13

Table 15.4-12

MSLBA Environment Releases

Isotope	Equilibrium Iodine Activity (MBq)	Pre-incident Spike (MBq)
Kr-85	2.0E+00	2.0E+00
Kr-85M	5.1E+02	5.1E+02
Kr-87	1.7E+03	1.7E+03
Kr-88	1.7E+03	1.7E+03
Xe-133	7.3E+02	7.3E+02
Xe-135	2.0E+03	2.0E+03
I-131	2.9E+05	5.8E+06
I-132	2.8E+06	5.5E+07
I-133	2.0E+06	4.0E+07
I-134	5.1E+06	1.0E+08
I-135	2.8E+06	5.7E+07

Table 15.4-13

MSLBA Analysis Results

Exposure Location and Time Period/Duration	**Maximum Calculated TEDE** (rem)	**Acceptance Criterion TEDE** (rem)
Exclusion Area Boundary (EAB) for the Entire Period of the Radioactive Cloud Passage		
Pre-incident Spike	7.6	25
Equilibrium Iodine Activity	0.5	2.5
Outer Boundary of Low Population Zone (LPZ) for the Entire Period of the Radioactive Cloud Passage		
Pre-incident Spike	7.6	25
Equilibrium Iodine Activity	0.5	2.5
Control Room Dose for the Duration of the Accident	4.7	5

Table 15.4-14

Feedwater Line Break Accident Parameters

I. Data and Assumptions Used to Estimate Source Terms	
A. Total mass of coolant released, kg (lb)	259,654 (571,925)
B. % of released coolant flashed to steam	22
C. Demineralizer efficiency, %	99
II. Data and Assumptions Used to Estimate Activity Released	
A. Iodine water concentration	0.2 μCi/g DE I-131
B. Iodine plateout fraction	0
C. Building release rate	Instantaneous
III. Dispersion and Dose Data	
A. Meteorology	1.00E-03 s/m^3
B. Method of dose calculation	Reference 15.4-1
C. Dose conversion assumptions	Reference 15.4-1, FGR-11
D. Activity inventory/releases	Table 15.4-15
E. Dose evaluations	Table 15.4-16

Table 15.4-15

Feedwater Line Break Accident

Environment Releases

Isotope	Activity (MBq)
I-131	1.3E+02
I-132	1.2E+03
I-133	8.7E+02
I-134	2.2E+03
I-135	1.2E+03

Table 15.4-16

Feedwater Line Break Analysis Results

Exposure Location and Time Period/Duration	Maximum Calculated TEDE (rem)	Acceptance Criterion TEDE (rem)
Exclusion Area Boundary (EAB) for the Entire Period of the Radioactive Cloud Passage	1.7E-04	2.5
Outer Boundary of Low Population Zone (LPZ) for the Entire Period of the Radioactive Cloud Passage	1.7E-04	2.5

Table 15.4-17

Instrument Line Break Accident Parameters

I. Data and assumptions used to estimate source terms		
A.	Power level, MWt	4590
B.	Mass of fluid released, kg (lbm)	14,785 (32,565)
C.	Mass of fluid flashed to steam, kg (lbm)	4,007 (8,825)
D.	Duration of accident, hr	6
E.	Number of bundles in core	1132
II. Data and assumptions used to estimate activity released		
A.	Iodine water concentration	0.2 μCi/g DE I-131
B.	Iodine Spiking	
	I-131, MBq/bundle (Ci/bundle)	7.77E+04 (2.1)
	I-132, MBq/bundle (Ci/bundle)	1.18E+05 (3.2)
	I-133, MBq/bundle (Ci/bundle)	1.85E+05 (5.0)
	I-134, MBq/bundle (Ci/bundle)	2.00E+05 (5.4)
	I-135, MBq/bundle (Ci/bundle)	1.78E+05 (4.8)
C.	Iodine plateout fraction, %	0
D.	Reactor Building Flow rate, %/hour	200
III Dispersion and Dose Data		
A.	Meteorology	3.00E-03 s/m^3
B.	Method of Dose Calculation	Ref 15.4-1
C.	Dose conversion Assumptions	RG 1.183,and Ref. 15.4-1
D.	Activity Inventory/releases	Table 15.4-18
E.	Dose evaluations	Table 15.4-19

Table 15.4-18

Instrument Line Break Accident Isotopic Inventory

ISOTOPE	1-min	10-min	1-hour	2-hour	4-hour	8-hour	12-hour
A. Reactor Building Inventory, MBq							
I-131	3.37E+03	2.91E+04	1.14E+05	2.04E+05	2.29E+05	1.28E+03	4.22E-01
I-132	6.88E+03	5.74E+04	1.71E+05	2.16E+05	1.41E+05	2.62E+02	2.63E-02
I-133	9.03E+03	7.77E+04	2.88E+05	4.81E+05	5.03E+05	2.49E+03	7.33E-01
I-134	1.18E+04	9.32E+04	2.05E+05	1.85E+05	7.29E+04	4.07E+01	5.77E-04
I-135	9.36E+03	7.99E+04	2.76E+05	4.22E+05	3.85E+05	1.44E+03	3.18E-01
TOTAL	4.03E+04	3.37E+05	1.05E+06	1.51E+06	1.33E+06	5.51E+03	1.50E+00
B. Reactor Building Inventory, Curies							
I-131	9.11E-02	7.86E-01	3.07E+00	5.51E+00	6.20E+00	3.46E-02	1.14E-05
I-132	1.86E-01	1.55E+00	4.62E+00	5.84E+00	3.80E+00	7.08E-03	7.12E-07
I-133	2.44E-01	2.10E+00	7.78E+00	1.30E+01	1.36E+01	6.74E-02	1.98E-05
I-134	3.20E-01	2.52E+00	5.54E+00	5.00E+00	1.97E+00	1.10E-03	1.56E-08
I-135	2.53E-01	2.16E+00	7.46E+00	1.14E+01	1.04E+01	3.89E-02	8.59E-06
TOTAL	1.09E+00	9.11E+00	2.85E+01	4.08E+01	3.60E+01	1.49E-01	4.05E-05
C. Isotopic Release to Environment, MBq							
I-131	2.15E+01	1.95E+03	5.00E+04	1.81E+05	3.96E+05	4.92E+05	4.92E+05
I-132	4.40E+01	3.89E+03	8.40E+04	2.38E+05	4.11E+05	4.55E+05	4.55E+05
I-133	5.77E+01	5.22E+03	1.29E+05	4.48E+05	9.32E+05	1.14E+06	1.14E+06
I-134	7.59E+01	6.40E+03	1.14E+05	2.60E+05	3.74E+05	3.96E+05	3.96E+05
I-135	5.99E+01	5.37E+03	1.28E+05	4.11E+05	8.10E+05	9.58E+05	9.58E+05
TOTAL	2.59E+02	2.28E+04	5.07E+05	1.54E+06	2.92E+06	3.44E+06	3.44E+06
D. Isotopic Release to Environment, Curies							
I-131	5.82E-04	5.27E-02	1.35E+00	4.90E+00	1.07E+01	1.33E+01	1.33E+01
I-132	1.19E-03	1.05E-01	2.27E+00	6.42E+00	1.11E+01	1.23E+01	1.23E+01
I-133	1.56E-03	1.41E-01	3.49E+00	1.21E+01	2.52E+01	3.09E+01	3.09E+01
I-134	2.05E-03	1.73E-01	3.09E+00	7.02E+00	1.01E+01	1.07E+01	1.07E+01
I-135	1.62E-03	1.45E-01	3.45E+00	1.11E+01	2.19E+01	2.59E+01	2.59E+01
TOTAL	7.00E-03	6.17E-01	1.37E+01	4.15E+01	7.89E+01	9.31E+01	9.31E+01

Table 15.4-19

Instrument Line Break Accident Results

Exposure Location and Time Period/Duration	**Maximum Calculated TEDE** (rem)	**Acceptance Criterion TEDE** (rem)
Exclusion Area Boundary (EAB) for the Entire Period of the Radioactive Cloud Passage	0.7	2.5
Outer Boundary of Low Population Zone (LPZ) for the Entire Period of the Radioactive Cloud Passage	0.7	2.5

Table 15.4-20

RWCU/SDC System Line Failure Outside Containment Sequence of Events

Sequence of Events	Elapsed Time
Clean up water line break occurs	0
Check valves on clean up water line to feedwater line isolate. Differential pressure instrumentation initiates delay sequence	0+
Differential pressure instrumentation actuates isolation valves	45 min
Isolation valves complete closure and isolation	75 min
Normal reactor shutdown and cooldown procedure	1-2 hour

Table 15.4-21

RWCU/SDS Line Break Accident Parameters

I. Data and assumptions used to estimate source terms		
A.	Power level, MWt	4590
B.	Number of bundles in core	1132
C.	Duration of accident, hr	< 2
II. Data and assumptions used to estimate activity released		
A.	Iodine water concentration	0.2 μCi/g DE I-131
B.	Iodine Spiking	
	I-131, MBq/bundle (Ci/bundle)	7.77E+04 (2.1)
	I-132, MBq/bundle (Ci/bundle)	1.18E+05 (3.2)
	I-133, MBq/bundle (Ci/bundle)	1.85E+05 (5.0)
	I-134, MBq/bundle (Ci/bundle)	2.00E+05 (5.4)
	I-135, MBq/bundle (Ci/bundle)	1.78E+05 (4.8)
C.	Iodine plateout fraction, %	0
D.	Reactor Building Flow rate, %/hour	Instantaneous
III Dispersion and Dose Data		
A.	Meteorology	1.00E-03 s/m^3
B.	Method of Dose Calculation	Ref 15.4-1
C.	Dose conversion Assumptions	RG 1.183,and Ref. 15.4-1
D.	Activity Inventory/releases	Table 15.4-22
E.	Dose evaluations	Table 15.4-23

Table 15.4-22

RWCU/SDS Line Break Accident

Isotopic Release to Environment

Isotope	Activity (MBq)
I-131	1.03E+06
I-132	2.09E+06
I-133	2.75E+06
I-134	3.61E+06
I-135	2.84E+06

Table 15.4-23

RWCU/SDS Line Break Accident Results

Exposure Location and Time Period/Duration	**Maximum Calculated TEDE** (rem)	**Acceptance Criterion TEDE** (rem)
Exclusion Area Boundary (EAB) for the Entire Period of the Radioactive Cloud Passage	0.6	2.5
Outer Boundary of Low Population Zone (LPZ) for the Entire Period of the Radioactive Cloud Passage	0.6	2.5

Table 15.4-24

Spent Fuel Cask Drop Accident Parameters

I	**Data and Assumptions Used to Estimate Source Terms**	
A.	Core power	4590 MWt
B	Burnup of fuel	58 GWd/MT
C.	Radial peaking factor	2.0
D.	Fuel bundles in cask	18
E.	Damaged fuel bundles	18
F.	Minimum time of fuel in storage prior to accident	120 days
G.	Time period for Reactor Building release	0 hr (Instantaneous)
H.	Fraction of activity released	10% of all isotopes, except 30% of Kr-85
I.	Iodine filter efficiency	None
II	**Dispersion and Dose Data**	
A.	Meteorology	Table 15.3-17
B.	Breathing rate	3.5E-04 m^3/s
C.	Dose conversion assumptions	RG 1.183
D.	Activity released	Table 15.3-16
E.	Dose consequences	Table 15.3-17

Table 15.4-25

Spent Fuel Cask Drop Accident

Isotopic Releases to Environment

Isotope	Activity (MBq)
I-131	4.7E+05
Kr-85	5.3E+08
Xe-131m	1.6E+05
Xe-133	4.4E+03

Table 15.4-26

Spent Fuel Cask Drop Accident Meteorology and Dose Results

Meteorology (s/m^3)	**Maximum Calculated TEDE** (rem)	**Acceptance Criterion TEDE** (rem)
5.0E-03	0.8	6.3

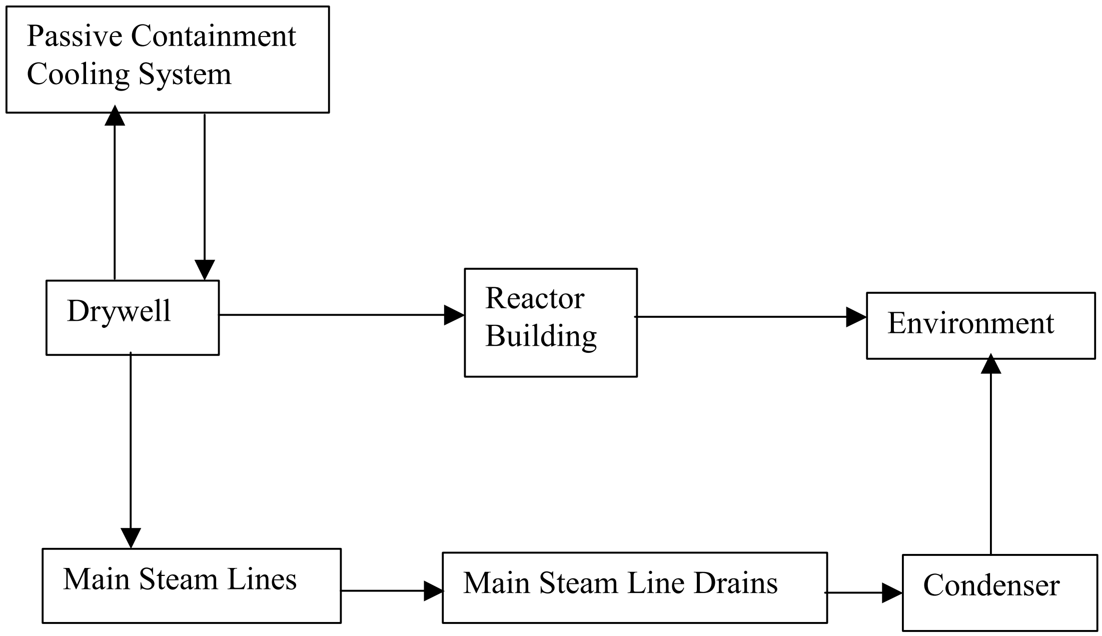


Figure 15.4-1. LOCA Radiological Paths

15.5 SPECIAL EVENT EVALUATIONS

15.5.1 Overpressure Protection

Results of the overpressure protection evaluation are provided in Subsection 5.2.2.

15.5.2 Shutdown Without Control Rods (Standby Liquid Control System Capability)

Reactor shutdown without control rods is an event requiring an alternate method of reactivity control—the Standby Liquid Control System (SLCS). The safety evaluation of SLCS capability is described in Subsection 9.3.5.3.

15.5.3 Shutdown from Outside Main Control Room

Reactor shutdown from outside the main control room is an event investigated to evaluate the capability of the plant to be safely shutdown and cooled to the cold shutdown state from outside the main control room. The evaluation is described in Subsection 7.4.2.

15.5.4 Anticipated Transients Without Scram

15.5.4.1 Requirements

NUREG-0800 Standard Review Plan (SRP) 15.8 requires the BWR to have an automatic recirculation pump trip (RPT) and emergency procedures for ATWS. This SRP has been superseded by the issuance of 10 CFR 50.62, which requires the BWR to have automatic RPT (not applicable to an ESBWR), an Alternate Rod Insertion (ARI) system, and an automatic SLCS. The SLCS is required to have a minimum flow capacity and boron content equivalent to 5.42×10^{-3} m^3/sec (86 gpm) of 13 weight percent sodium pentaborate solution.

15.5.4.2 Plant Capabilities

For ATWS prevention/mitigation, the ESBWR provides the following:

- An ARI system that utilizes sensors and logic that are diverse and independent of the RPS;
- Electrical insertion of Fine Motion Control Rod Drives (FMCRDs) that also utilize sensors and logic that are diverse and independent of the RPS;
- Automatic feedwater runback under conditions indicative of an ATWS; and
- Automatic initiation of SLC under conditions indicative of an ATWS.

The ATWS rule of 10 CFR 50.62 was written as hardware-specific, rather than functionally, because it clearly reflected the BWR use of forced core flow circulation. Because the ESBWR uses natural circulation, there are no recirculation pumps to be tripped. Hence, no RPT logic can be implemented in the ESBWR. An ATWS automatic feedwater runback feature is implemented, to provide a reduction in water level, core flow and reactor power, similar to RPT in a forced circulation plant. This feature prevents reactor vessel overpressure and possible short-term fuel damage for the most limiting ATWS events.

The ATWS rule of 10 CFR 50.62 is also specific to the use of locking-piston control rod drives. The ESBWR, however, uses the FMCRD design with both hydraulic and electrical means to achieve shutdown. This drive system is described in detail in Section 4.6. The use of this design eliminates the common mode failure potentials of the existing locking-piston CRD by eliminating the scram discharge volume (potential mechanical common mode failure) and by having an electric motor run-in diverse from the hydraulic scram feature. This latter feature allows rod run-in, if scram air header pressure is not exhausted because of a postulated common mode electrical failure and simultaneous failure of the ARI system, and thus satisfies the intent of 10 CFR 50.62. Therefore, the ESBWR design can respond to an ATWS threatening event independent of SLCS.

The SLCS is required by 10 CFR 50 Appendix A, and is described within Section 9.3. Because the new drive design eliminates the previous common-mode failure potential and because of the very low probability of simultaneous common mode failure of a large number of FMCRDs, a failure to achieve shutdown is deemed incredible. However, automatic initiation of the SLCS under conditions indicative of an ATWS is also incorporated in order to meet the rule specified in 10 CFR 50.62.

15.5.4.3 Performance Evaluation

15.5.4.3.1 Introduction

Typical ATWS events are analyzed to confirm the design for ESBWR.

The procedure and assumptions used in this analysis are consistent with those used in the analyses for the operating plants as documented in Reference 15.5-1.

All transient analyses, unless otherwise specified, were performed with the TRACG code.

15.5.4.3.2 Performance Requirements

As identified in Reference 15.5-1, the design should meet the following requirements:

Fuel Integrity—The long-term core cooling capability shall be assured by meeting the cladding temperature and oxidation criteria of 10 CFR 50.46 [i.e., peak cladding temperature (PCT) not exceeding 1204°C (2200°F), and the local oxidation of the cladding not exceeding 17% of the total cladding thickness].

Containment Integrity—The long-term containment capability shall be maintained. The maximum containment pressure shall not exceed the design pressure of the containment structure, 414 KPa (60 psia). The suppression pool temperature shall be limited to the wetwell design temperature of 121°C (250°F).

Primary System—The system transient pressure shall be limited such that the maximum primary stress within the reactor coolant pressure boundary (RCPB) does not exceed the emergency limits as defined in the ASME Code, Section III.

Long-Term Shutdown Cooling—Subsequent to an ATWS event, the reactor shall be brought to a safe shutdown condition, and be cooled down and maintained in a cold shutdown condition.

These performance requirements are summarized in Table 15.5-1.

15.5.4.3.3 Analysis Conditions

The probability of the occurrence of an ATWS is low. Thus, historically nominal parameters and initial conditions have been used in these analyses as specified in Reference 15.5-1.

As the processes for definition of allowable operational flexibility and margin improvement options expanded (see Section S.5 of Reference 15.5-2), the analysis process transitioned to a basis that required use of bounding initial conditions. This was done because the frequency of operation within the allowable optional configurations could not be defined. In other words, "nominal" could not be defined. Some initial conditions, the most important being reactor power, are still analyzed without consideration of instrument uncertainties. Those that are applied conservatively include core exposure, core axial power shape, and Safety/Relief Valve operability. All events analyzed assume reduced IC heat removal capacity to add a further measure of conservatism. The peak containment pressure presented is estimated in a conservative manner assuming that all the non-condensable gas from the drywell is in the wetwell airspace at the time of the peak pool temperature. Credit has been taken for the high-pressure control rod drive (HPCRD) flow in all the cases analyzed.

Nominal plant parameters are also difficult to define. Processes for definition of bounding "analytical limits" are much clearer and much more defensible than are nominal parameters. For this reason, ATWS analytical parameters have tended toward application of bounding parameters for selected plant parameters that affect the critical safety parameters. The most important parameters for peak vessel pressure are Safety/Relief Valve capacities and setpoints. The most important parameters for clad and suppression pool temperature are initial Critical Power Ratio and boron flow rate.

Tables 15.5-2 and 15.5-3 list the initial conditions and equipment performance characteristics, which are used in the analysis.

15.5.4.3.4 ATWS Logic and Setpoints

The mitigation of ATWS events is accomplished by a multitude of equipment and procedures. These include ARI, FMCRD run-in, feedwater runback, ADS inhibit, and SLC. The logic of this ATWS mitigation logic is presented in Figure 7.3-5 and Subsections 7.7.2 and 7.3.4. The following are the initiation signals and setpoints for the above response:

- ARI and FMCRD run-in
  - High pressure, or
  - Level 2, or
  - Manual.
- SLCS initiation
  - High pressure and average power range monitoring (APRM) not downscale for 3 minutes, or
  - Level 2 and APRM not downscale for 3 minutes, or
  - Manual ARI/FMCRD run-in signals and APRM not downscale for 3 minutes.

- Feedwater runback
  - High pressure and APRM not downscale, or
  - Manual ARI/FMCRD run-in.
- ADS inhibit
  - High pressure and APRM not downscale for one minute, or
  - Level 2 and APRM not downscale.

15.5.4.3.5 Selection of Events

Based on conclusions from the evaluations for operating BWR plants as documented in Reference 15.5-1, events were selected to demonstrate the performance of the ATWS capabilities. The events are grouped into three categories. The first category includes events that demonstrate ATWS mitigation on the most severe and limiting cases. The second category has events that are generally less severe for ATWS analysis but are analyzed to show the sensitivity of key ATWS parameters to these events. In each of the above cases, ATWS mitigation actions are assumed to occur on the appropriate signals. No operator action is assumed, unless specifically mentioned. The third category covers the cases that have only minor effect on the reactor vessel containment. They are discussed briefly to support the assumption that they do not significantly influence the design of ATWS mitigation. No analysis was performed for events in the third category.

Category 1: Limiting Events

- Main Steamline Isolation Valve (MSIV) Closure - Generic studies have shown that this transient produces high neutron flux, vessel pressure, and suppression pool temperature. The maximum values from this event are, in most cases, bounding of all events considered.
- Loss of Condenser Vacuum - The turbine will trip on low condenser vacuum. The bypass valves are available for a short period, and then close on loss of condenser vacuum. Depending on detailed BOP performance the pressurization rate and the energy addition to the pool may be as severe as MSIVC. This event is included in category I to assure the short term peak vessel pressure and clad temperature remain within limits.
- Loss of Feedwater Heating - In ESBWR this event is mitigated with Select Control Rod Run-In (SCRRI). Consistent with ATWS failure to scram, this event will be evaluated with no SCRRI. This event is included in category I, to determine whether it will be limiting for peak clad temperature. Because the turbine bypass valves are available, it is not limiting for vessel pressure or suppression pool temperature.

Category 2: Moderate Events

- Loss of Normal AC Power to Station Auxiliaries - This transient is less severe than the MSIV closure in terms of vessel pressure, neutron flux, and suppression pool temperature. However, because of the loss of AC power, the availability of equipment is different. Therefore, the plant capability of mitigating this event is evaluated.

- Loss of Feedwater Flow - This transient is less severe than the above events. However, it is the only event where the ATWS trip is initiated from the low level signals. Thus, this event is analyzed to show that the low level trips are capable to mitigate the event.
- Generator Load Rejection with a Single Failure in the Turbine Bypass System – In this transient, because half of the bypass valves are available, the pressurization rate is less severe than MSIVC, the FW temperature change will be similar to MSIVC and the energy addition to the pool is less severe than MSIVC.

Category 3: Minimum Effect Events

- Inadvertent Isolation Condenser Initiation - Spurious initiation of the isolation condensers would cause a moderator temperature decrease and a slow insertion of positive reactivity into the core. During power operation the system will settle at a new steady state.
- Turbine Trip with Full Bypass – In this transient, because full bypass capacity is available, the pressurization rate is less severe than MSIVC, the FW temperature change will be similar to MSIVC and the energy addition to the pool is less severe than MSIVC.
- Opening of One Control or Turbine Bypass Valve – This event assumes a hydraulic system failure that causes a mild decrease in pressure, which is compensated for by the control system closing other valves. The ATWS response will not be limiting.

15.5.4.3.6 Transient Responses

Three cases are analyzed for the limiting transient-MSIV closure. The first one shows the ATWS performance with ARI. This case is intended to show the effectiveness of the ARI design. The second case, which uses FMCRD run-in, assuming a total failure of ARI, was performed to show the backup capability of FMCRD run-in. The third case is analyzed to show the in-depth ATWS mitigation capability of the ESBWR. In this case, both ARI and FMCRD run-in are assumed to fail. Automatic boron injection with a 180-second delay is relied upon to mitigate the transient event. For the other events analyzed, only the ATWS mitigation capability with boron injection is examined.

If the ARI and FMCRD run-in fail at the same time, which has extremely low probability of occurrence, the peak reactor pressure would still be controlled by the recirculation runback and SRVs. However, the nuclear shutdown then relies on the automatic SLCS injection. The boron would reach the core in about 5 seconds after the initiation. The operation of accumulator driven SLCS produces the initial volumetric flow rate of sodium pentaborate shown in Table 15.5-2. The nuclear shutdown would begin when boron reaches the core.

Stability performance during an ATWS event is examined for the MSIV closure case and the results are discussed at the end of this section.

Main Steam Isolation Valve Closure

This transient is considered an initiating event caused by either operator action or instrument failure. Scram signal paths that are assumed to fail include valve position, high neutron flux, high vessel pressure, and all manual attempts. A short time after the MSIVs have closed completely, the ATWS high pressure setpoint is reached, which initiates the actuation of ARI and FMCRD run-in. The insertion of the control rods is successful in bringing the reactor to hot shutdown. Peak values of key parameters are shown in Table 15.5-4a for the ARI case and

Table 15.5-4b for the FMCRD run-in case. In the case that control rods fail to insert, the reactor would be brought to hot shutdown by automatic SLC boron injection. Operator actions during this event include controlling the water level to maintain a minimum of 1.5m (5') above the top of active fuel (TAF) and depressurizing via the SRVs to maintain margin to the Heat Capacity Temperature Limit (HCTL) for the suppression pool. The transient behavior of this case is listed in Table 15.5-4c. A sequence of the main events that occur during this transient is presented in Table 15.5-4d. The reactor system responses are presented in Figures 15.5-1a-d for the ARI case, Figures 15.5-2a-d for the FMCRD run-in case and Figures 15.5-3a-d for the SLCS case, respectively.

Loss of Condenser Vacuum

This transient starts with a turbine trip because of the low condenser vacuum; therefore, the beginning is the same as the turbine trip event. However, the MSIVs and turbine bypass valves also close after the condenser vacuum has further dropped to their closure setpoints. Hence, this event is similar to the MSIV closure event for all the key parameters. Transient behavior is shown in Figures 15.5-4a-d for the SLC case. The high pressure ATWS setpoint is reached shortly after the closure of MSIV. The high pressure initiates ARI, FMCRD run-in and the SLC timer. The SLCS trip is activated upon APRM not downscale and high-pressure signals and boron flow starts 3 minutes following the trip. As the poison reaches sufficient concentration in the core, the reactor achieves hot shutdown. Table 15.5-5a shows the summary of peak values of key parameters for this event and Table 15.5-5b presents a sequence of main events that occur during this transient.

Loss of Feedwater Heater

This transient does not trip any automatic ATWS logic. It is assumed that the operator pushes the ARI buttons at approximately 10 minutes after the beginning of this event. This action initiates ARI, FMCRD run-in, and SLC timer. At this time, the reactor has settled in a new steady state at a higher power level. However, the feedwater runback initiated by manual ARI signal and APRM not-downscale signal causes the water level to drop below Level 2. Low water level results in a closure of all MSIVs, and subsequent reactor pressure increase. The pressure increase is mitigated by SRV opening. Upon failure of rod insertion, SLC can bring the reactor to hot shutdown at approximately 15 minutes after the event starts. The transient behavior for the case is shown in Figure 15.5-5a-d. The peak values of the key parameters are shown in Table 15.5-6a. Table 15.5-6b presents a sequence of main events that occur during this transient.

Loss of Non-Emergency AC Power to Station Auxiliaries

In this event, all scram signal paths, including valve position, high flux, high pressure, low level, and all manual attempts have been assumed to fail.

The loss of AC power has the following effects:

- An immediate load rejection occurs. This causes fast closure of the turbine control valves.
- Due to the loss of power to the condensate and feedwater pumps, feedwater is lost.
- The reactor is isolated after loss of main condenser vacuum.

Figures 15.5-6a-d show the transient behavior for the case with automatic SLCS initiation.

The fast closure of the turbine control valves causes a rapid increase of pressure, and the ATWS high pressure setpoint is reached shortly after the control valves have closed. The ATWS high-pressure signal initiates ARI and FMCRD run-in. If both modes of rod insertion fail, the ATWS high-pressure signal also initiates the timer for SLC. After confirming the rod insertion failure by monitoring the high pressure and APRM not-downscale signal for 3 minutes, the SLCS would be initiated. The reactor is brought to hot shutdown when enough boron concentration is built up in the reactor core.

Table 15.5-7a shows the summary of peak values of key parameters for the event. Table 15.5-7b presents a sequence of main events that occur during this transient.

Loss of Feedwater Flow

This event does not have rapid excursions, as in some of the other events, but is a long-term power reduction and depressurization. Because the pressure begins to fall at the onset of the transient, SRVs are not required until isolation occurs very late in the event and only single group valve cycling is expected to handle decay heat. The containment limits are not approached.

In this event all feedwater flow is assumed to be lost in about five seconds. The mitigation of this event by the SLCS is illustrated in Figures 15.5-7a-d.

After the loss of feedwater has taken place, the pressure, water level and neutron flux begin to fall. Reaching low water Level 2 (L2) activates ARI and FMCRD run-in and starts the SLCS timer, closes MSIVs, and initiates CRD high pressure make-up and isolation condensers. Failure of rod insertion initiates SLC when the timer times out while the APRM signal is not downscale. The reactor reaches the hot shutdown condition as the boron concentration builds up the core. Table 15.5-8a shows the summary of peak values of key parameters for the case. Table 15.5-8b presents a sequence of main events that occur during this transient.

Load Rejection with a Single Failure in the Turbine Bypass System

The initial characteristics of this transient are much like the MSIV closure described above with a rapid steam shutoff. Pressure and power increases are limited by the action of the SRVs and feedwater runback. As this event progresses, however, the availability of the main condenser makes it possible for the SRVs to be closed sooner and terminates the steam discharge to the suppression pool. The mitigation of this event with the SLCS is illustrated in Figures 15.5-8a-d.

The closure of the turbine control valves causes a rapid increase of pressure. The ATWS high-pressure setpoint is reached shortly after the closure. The high pressure initiates ARI, FMCRD run-in and the SLC timer. If the rods fail to insert into the core, the SLCS is initiated by the APRM signal not downscale and the high-pressure signal when the timer times out. Table 15.5-9a shows the summary of peak values of key parameters for these events. Table 15.5-9b presents a sequence of main events that occur during this transient.

Stability during ATWS

Studies are performed to examine core stability during ATWS. Perturbations are introduced in the core power at different times during the transient when power-flow ratios are steady and high, and the transient response to these perturbations is evaluated. The limiting case, where a perturbation was introduced at 25s, is illustrated in Figure 15.5-9. The perturbations in the power are quickly damped out, indicating that the ESBWR operation remains stable during these events.

15.5.4.4 Conclusion

Based upon the results of this analysis, the proposed ATWS design for the ESBWR is satisfactory in mitigating the consequences of an ATWS. All performance requirements specified in Subsection 15.5.4.3.2 are met. It is also demonstrated that the plant operation remains stable during an ATWS event.

It is also concluded from results of the above analysis that automatic boron injection could mitigate the most limiting ATWS event with margin. Therefore, an automatic SLCS injection as a backup for ATWS mitigation is acceptable.

15.5.5 Station Blackout

The performance evaluation for Station Blackout (SBO) based on TRACG to the requirements of 10 CFR 50.63 is presented in this subsection.

15.5.5.1 Acceptance Criteria

The design shall meet the following acceptance criteria:

- **Reactor Vessel Coolant Integrity** - Adequate reactor coolant inventory shall be maintained such that reactor water level is maintained above the core (i.e., top of active fuel).
- **Hot Shutdown Condition** - Achieve and maintain the plant to those shutdown conditions specified in plant Technical Specifications as Hot Shutdown.
- **Containment Integrity** – If containment isolation is involved, the maximum containment and suppression pool pressures and temperatures shall be maintained below their design limits.

15.5.5.2 Analysis Assumptions

The analysis assumptions and inputs are summarized below.

- Reactor is operating initially at 100% of rated power/100% rated nominal core flow, nominal dome pressure and normal water level.
- The nominal ANSI/ANS 5.1-1994 decay heat model is assumed.
- SBO starts with loss of all alternating current (AC) power, which occurs at time zero.
- Loss of AC power trips reactor feedwater pumps at event initiation.
- The reactor scrams occurs at 2.0 seconds due to loss of power supply to feedwater pumps. When feedwater flow is lost, there is a scram signal with a delay time of 2.0 seconds.
- Feedwater is ramped down linearly to zero in 5.0 seconds after event initiation.

- Closure of all Main Steam Isolation Valves (MSIVs) is automatically initiated when the reactor water level reaches Level 2, and the valves are fully closed at 5.0 seconds.
- CRD pumps are not available due to loss of all AC power. The systems available for initial vessel inventory and pressure control, containment pressure/temperature control and suppression pool temperature control are:
  - Isolation Condensers (ICs)
  - Standby Liquid Control (SLC) system
  - Safety Relief Valves (SRVs)
  - Depressurization Valves (DPVs)
  - Gravity-Driven Cooling System (GDCS) squib valves
  - GDCS loops
  - Passive Containment Cooling System (PCCS) loops.
- The passive IC system is automatically initiated upon the closure of MSIVs to remove decay heat following scram and isolation and IC drain flow provides initial reactor coolant inventory markup to the reactor pressure vessel.
- When the water level reaches Level 1.5 and high drywell pressure setpoint, the Automatic Depressurization System (ADS) with operation of SRVs and DPVs is initiated to depressurize the reactor pressure vessel.
- The SLC system starts injection to provide additional reactor coolant inventory makeup after DPVs open and the water level reaches Level 1.
- After ADS initiation, GDCS loops automatically function as designed to provide more vessel coolant inventory makeup to restore the water level.
- PCCS heat exchanges condense steam to limit containment pressure to less than its design pressure.
- At the end of 8-hour coping period, the event is terminated.

15.5.5.3 Analysis Results

As shown in Figures 15.5-10 and 15.5-11, with operation of ADS, GDCS, SLC and IC systems, the reactor water levels in the core and the downcomer are maintained above the top of active fuel. Therefore the requirement for reactor vessel coolant integrity is satisfied.

Subsequent to a SBO event, hot shutdown condition can be achieved and maintained by operation of ADS, GDCS, SLC and IC systems. Therefore, the requirement for achieving and maintaining hot shutdown condition is met.

With operation of PCCS and IC system, the containment and suppression pool pressures and temperatures are maintained within their design limits as shown in Figures 15.5-12 and 15.5-13. Therefore, the integrity for containment is maintained.

As demonstrated above, each acceptance criterion in Subsection 15.5.5.1 is met. Therefore ESBWR can successfully mitigate a SBO event to meet the requirements of 10 CFR 50.63.

15.5.6 Safe Shutdown Fire

The fire hazard analysis is provided in Appendix 9A. The performance evaluation based on TRACG evaluation of loss of FW and loss of AC power/auxiliary transformer is presented in this subsection.

15.5.6.1 Acceptance Criteria

The design shall meet the following acceptance criteria:

- **Core Subcriticality** - Core subcriticality shall be achieved and maintained with adequate core shutdown margin, as specified in the plant Technical Specifications.
- **Reactor Vessel Coolant Integrity** - Adequate reactor coolant inventory shall be maintained such that reactor water level is maintained above the core (i.e., top of active fuel).
- **Hot Shutdown Condition** - Hot shutdown conditions shall be achieved and maintained.
- **Cold Shutdown Condition** - The cold shutdown condition shall be achieved within 72 hours and maintained thereafter.
- **Containment Integrity** – If containment isolation is involved, the maximum containment and suppression pool pressures and temperatures shall be maintained below their design limits.

15.5.6.2 Analysis Assumptions

The worst fire scenario is a fire occurring in the main control room (MCR), and requires operator evacuation. The analysis assumptions and inputs are summarized below.

- Reactor is operating initially at 100% of rated power/100% rated nominal core flow, nominal dome pressure and normal water level.
- The nominal ANSI/ANS 5.1-1994 decay heat model is assumed.
- Order to evacuate the MCR due to a MCR fire and loss of offsite power (LOOP) occurs at time zero.
- The reactor operator manually scrams the reactor before leaving the MCR.
- Closure of all Main Steamline Isolation Valves (MSIVs) is automatically initiated when the reactor water level reaches Level 2, and the valves are fully closed at 5 seconds.
- Feedwater flow is ramped down linearly to zero in 5 seconds after event initiation due to LOOP.
- A single failure is not assumed because fire protection does not require considering a single failure. The systems available for vessel inventory and pressure control, containment pressure/temperature control and suppression pool temperature control are:
  - Isolation Condensers (ICs)
  - Control Rod Drive (CRD) pumps
  - Fuel and Auxiliary Pools Cooling System (FAPCS) in any mode

- Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) in any mode
- Safety Relief Valves (SRVs)
- Depressurization Valves (DPVs)
- Gravity-Driven Cooling System (GDCS) squib valves
- GDCS loops
- Passive Containment Cooling System (PCCS) loops.

- No Spurious operation of SRV or DPV is assumed.
- It is conservatively assumed that it would take 10 minutes for operators to evacuate from MCR to remote shutdown panel (RSP).
- 4 ICs are automatically initiated upon the closure of MSIVs to stabilize the plant. Operators can monitor from RSP and manually control ICs to assure the maximum cooldown rate not exceeding 100^{o}F/hr, if necessary.
- When the reactor water level reaches Level 2, CRD pumps are automatically initiated to provide vessel inventory makeup. The maximum delayed time is 145 seconds upon restoring alternating current (AC) power because off-site power is not available. CRD pumps shall keep the water level above Level 0.5 to avoid any Automatic Depressurization System (ADS) initiation to blow down the reactor pressure vessel.
- After the operator regains the control in RSP, monitoring and manual control are necessary. RWCU/SDC shall be initiated following the normal shutdown procedure to ensure the reactor pressure vessel temperature is below 212^{o}F within 72 hours to meet the cold shutdown requirement.
- ICs and CRD pump flow stabilize the plant. SRVs, DPVs, PCCS and GDCS can be utilized if IC does not stabilize the plant, which is very unlikely.

15.5.6.3 Analysis Results

At event initiation, reactor scram occurs. Therefore core subcriticality is achieved and maintained.

The similar analysis results (a station blackout event) in Appendix 8B can be conservatively applied for this fire protection analysis, because more ICs are available for fire protection. As shown in Figure 8B-1, with operation of ICs and CRD pumps, the minimum reactor water level is about 4.67 m above the top of active fuel (about 12.12 m above vessel zero). The water level recovers above Level 1 lower analytical limit and upper analytical limit within approximately 5 minutes and 20 minutes into the event, respectively. Therefore the requirement for reactor vessel coolant integrity is satisfied. Additionally this minimum water level is well above Level 0.5, which is 11 m above vessel zero, and thus, ADS initiation can be avoided.

Subsequent to a fire event, hot shutdown condition can be achieved and maintained by operation of ICs and CRD pumps.

After reactor control room operators regain the control of the reactor in RSP, cold shutdown conditions can be achieved within 72 hours and maintained thereafter following the normal shutdown procedure because the control panel in RSP is identical to the one in MCR and the systems can be fully functioned as designed and can be controlled from RSP as in MCR.

ICs and CRD pumps stabilize the plant without SRV actuation or ADS blowdown, consequently there is no heat-up in the suppression pool and containment. Therefore, the integrity for containment is maintained.

As demonstrated above, each acceptance criterion in Subsection 15.5.6.1 is met. Therefore ESBWR can successfully mitigate a fire event.

15.5.7 Waste Gas System Leak or Failure

The safety analysis of waste gas system leak or failure is provided in Subsection 11.3.7.

15.5.8 References

15.5-1 General Electric Company, "Assessment of BWR Mitigation of ATWS," NEDE-24222, September 1979.

15.5-2 General Electric Company, "General Electric Standard Application for Reactor Fuel (GESTAR II)", NEDE-24011-P-A-14, June 2000.

Table 15.5-1

ATWS Performance Requirements

Case	**RPV Peak Pressure** MPa (psia)	**Maximum Pool Temperature** °C (°F)	**Fuel Integrity**	**Maximum Containment Pressure** kPa (psia)
ARI	10.34 (1500)	121(250)	Coolable Geometry	414(60)
FMCRD Run-in	10.34 (1500)	121(250)	Coolable Geometry	414(60)
Boron Injection	10.34 (1500)	121(250)	Coolable Geometry	414(60)

Table 15.5-2

ATWS Initial Operating Conditions

Parameters	Value
Dome Pressure, MPaG (psig)	7.07(1025)
Natural Circulation Core Flow, Mkg/hr (Mlb/hr)	36.9(81.4)
Vessel Diameter, m (ft)	7.1(23.3)
Numbers of Fuel Bundles	1132
Power, MWt/% NBR	4500/100
Steam/Feed Flow, kg/sec (Mlbm/hr)	2433(19.31)
Feedwater Temperature, °C (°F)	215.6(420)
Nuclear Condition	EOC
Suppression Pool Volume, m^3 (ft^3)	3610 (127,500)
Initial Suppression Pool Temperature, °C (°F)	43.3(110)
SLCS accumulator driven initial flow, m^3/s (gpm)	0.03 (475)

Table 15.5-3

ATWS Equipment Performance Characteristics

Parameters	Value
MSIV Closure Time, sec	≥3.0
Delay before start of Electro-Hydraulic Rod Insertion, sec	≤1
Electro-Hydraulic Control Rod Insertion Time, sec	≤130
Maximum time for start of motion of ARI rods, sec	15
Maximum time for all ARI rods to be fully inserted, sec	25
Safety/Relief Valve (SRV) System Capacity, % NBR Steam Flow/No. of Valves	≥89.5/18
SRV Setpoint Range, MPaG (psig)	8.618 to 8.756 (1250-1270)
SRV Opening Time, sec	<1.7
Pressure Drop Below Setpoint for SRV Closure [nominal/analysis assumption],% nameplate	≤96
CRD (high Pressure Make-Up Function) Low Water Level Initiation Setpoint, cm (in)	1605.0(631.9)
CRD (High Pressure Make-Up Function) Flow Rate, m^3/sec (gal/min)	0.065(1035)
ATWS Dome Pressure Sensor Time Constant, sec	≤0.5
ATWS Logic Time Delay, sec	≤1
Pool Cooling Capacity, MW	5.985
Low Water Level For Closure of MSIVs, cm (in)	1605.0(631.9)
Low Steamline Pressure For Closure of MSIVs, MPaG (psig)	5.412 (785)
Temperature For Automatic Pool Cooling, ºC (ºF)	48.9 (120)

Table 15.5-4a

ATWS MSIV Closure Summary - ARI Case

Parameter	Value	Time
Maximum Neutron Flux, %	228	3s
Maximum Vessel Bottom Pressure, MPaG (psig)	9.67 (1402.8)	20s
Maximum Bulk Suppression Pool Temperature, ºC (ºF)	54.0 (129.1)	37s
Associated Containment Pressure, MPaG (psig)	0.171 (24.81)	37s
Peak Cladding Temperature, ºC (ºF)	940.6 (1725)	25s

Table 15.5-4b

ATWS MSIV Closure Summary - FMCRD Case

Parameter	Value	Time
Maximum Neutron Flux, %	228	3s
Maximum Vessel Bottom Pressure, MPaG (psig)	9.83 (1426.1)	31s
Maximum Bulk Suppression Pool Temperature, ºC (ºF)	68.8 (155.8)	320s
Associated Containment Pressure, MPaG (psig)	0.197 (28.61)	320s
Peak Cladding Temperature, ºC (ºF)	952.8 (1747.0)	31s

Table 15.5-4c

ATWS MSIV Closure Summary - SLCS Case

Parameter	Value	Time
Maximum Neutron Flux, %	228	3s
Maximum Vessel Bottom Pressure, MPaG (psig)	9.81 (1422)	28s
Maximum Bulk Suppression Pool Temperature, ºC (ºF)	77.6(172)	254s
Associated Containment Pressure, MPaG (psig)	0.218(31.57)	254s
Peak Cladding Temperature, ºC (ºF)	916.2(1681.1)	25s

Table 15.5-4d

ATWS MSIVC Sequence of Events

Time (s)			Event
ARI	FMCRD	SLCS	
0	0	0	MSIV Closure starts
2	2	2	IC flow starts
4	4	34	ATWS trip set at high pressure
5	5	5	SRVs open
33	41	42	Level drops below L2 set point
43	51	52	HPCRD flow starts
-	-	189	SLCS injection starts*
-	-	710	Hot Standby Volume of Boron injected into Bypass
30	-	-	ARI fully inserted**
-	135	-	FMCRD Run-in complete***

* SLCS is automatically initiated if ARI and FMCRD functions fail
** ARI constitutes the first course of action for ATWS mitigation
*** FMCRD run-in for ATWS mitigation occurs if ARI fails

Table 15.5-5a

ATWS Loss of Condenser Vacuum Summary - SLCS Case

Parameter	Value	Time
Maximum Neutron Flux, %	218	3s
Maximum Vessel Bottom Pressure, MPaG (psig)	9.84 (1428)	29s
Maximum Bulk Suppression Pool Temperature, ºC (ºF)	80.6(177)	254s
Associated Containment Pressure, MPaG (psig)	0.225(32.70)	254s
Peak Cladding Temperature, ºC (ºF)	916.0(1681)	25s

Table 15.5-5b

ATWS Loss of Condenser Vacuum Sequence of Events

Time (s)	Event
0	Loss of Condenser Vacuum
0	Turbine Trip initiated
6	MSIV starts to close
10	ATWS trip set at high pressure
10	IC flow starts
11	SRVs open
49	Level drops below L2 set point
59	HPCRD flow starts
195	SLCS injection starts
716	Hot Standby Volume of Boron injected into Bypass

Table 15.5-6a

ATWS Loss of Feedwater Heating Summary - SLCS Case

Parameter	Value	Time
Maximum Neutron Flux, %	121	596s
Maximum Vessel Bottom Pressure, MPaG (psig)	8.62(1251)	724s
Maximum Bulk Suppression Pool Temperature, ºC (ºF)	50.2(122.3)	857s
Associated Containment Pressure, MPaG (psig)	0.166(24.01)	857s
Peak Cladding Temperature, ºC (ºF)	316.1(600.9)	620s

Table 15.5-6b

ATWS Loss of Feedwater Heating Sequence of Events

Time (s)	Event
0	Loss of Feedwater heating
600	Feedwater runback initiated
637	L2 setpoint reached
637	ATWS trip set at L2
638	MSIV closure starts
648	HPCRD flow starts
668	IC flow starts
692	SRVs open
785	SLCS flow starts
1300	Hot Standby Volume of Boron injected into Bypass

Table 15.5-7a

ATWS Loss of Non-Emergency AC Power to Station Auxiliaries Summary - SLCS Case

Parameter	Value	Time
Maximum Neutron Flux, %	183	12s
Maximum Vessel Bottom Pressure, MPaG (psig)	9.15 (1326.7)	15s
Maximum Bulk Suppression Pool Temperature, ºC (ºF)	70.3 (158.6)	324s
Associated Containment Pressure, MPaG (psig)	0.201 (29.09)	324s
Peak Cladding Temperature, ºC (ºF)	765.8 (1410.4)	18s

Table 15.5-7b

ATWS Loss of Non-Emergency AC Power to Station Auxiliaries Sequence of Events

Time (s)	Event
0	Loss of AC Power
8	MSIV Closure starts
10	IC flow starts
12	ATWS trip set at high pressure
14	SRVs open
32	Level drops below L2 set point
61	Level drops below L1 set point
152	HPCRD flow starts
197	SLCS injection starts
718	Hot Standby Volume of Boron injected into Bypass

Table 15.5-8a

ATWS Loss of Feedwater Flow Summary - SLCS Case

Parameter	Value	Time
Maximum Neutron Flux, %	100	0s
Maximum Vessel Bottom Pressure, MPaG (psig)	8.62 (1250)	95s
Maximum Bulk Suppression Pool Temperature, ºC (ºF)	52.1 (125.8)	327s
Associated Containment Pressure, MPaG (psig)	0.168(24.42)	327s
Peak Cladding Temperature, ºC (ºF)	314.1 (597.4)	0.2s

Table 15.5-8b

ATWS Loss of Feedwater Flow Sequence of Events

Time (s)	Event
0	Feedwater Pump coastdown starts
33	Level drops below L2 set point, ATWS trip is set
43	HPCRD flow starts
63	MSIV Closure starts
64	Level drops below L1 set point, IC flow starts
94	SRVs open
216	SLCS injection starts
739	Hot Standby Volume of Boron injected into Bypass

Table 15.5-9a

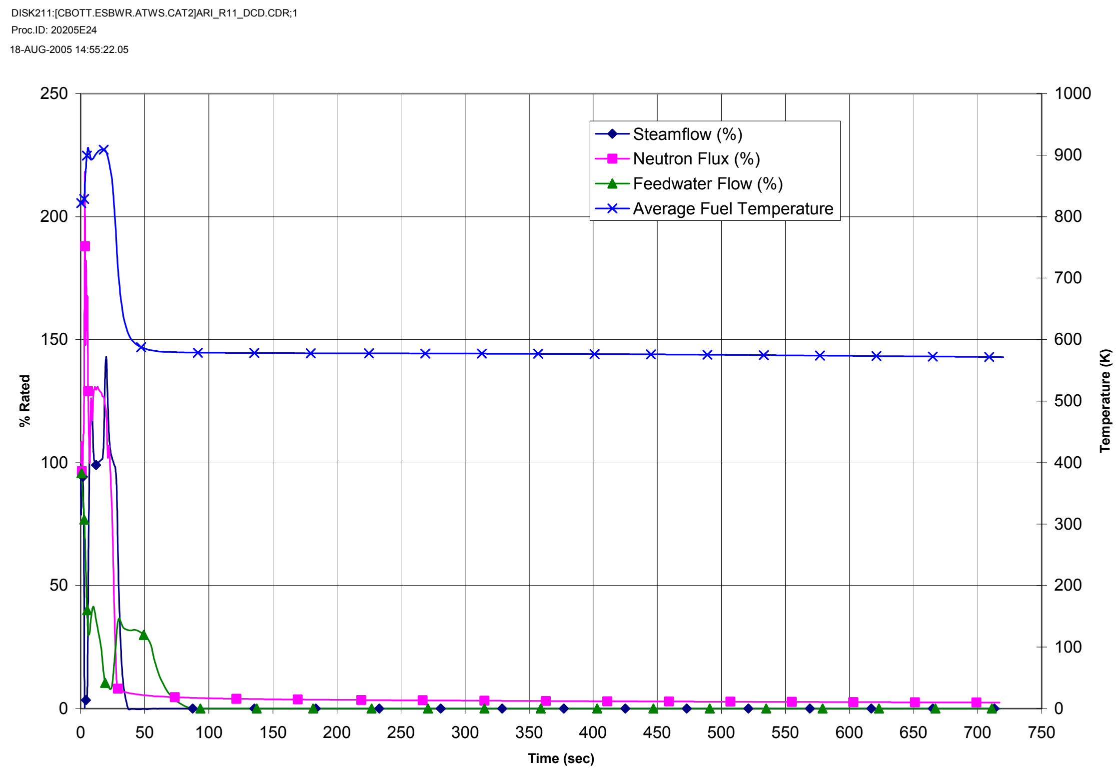
ATWS Load Rejection with a Single Failure in the Turbine Bypass System Summary - SLCS Case

Parameter	Value	Time
Maximum Neutron Flux, %	189	0.7s
Maximum Vessel Bottom Pressure, MPaG (psig)	8.82 (1279.0)	15s
Maximum Bulk Suppression Pool Temperature, ºC (ºF)	54.8 (130.6)	305s
Associated Containment Pressure, MPaG (psig)	0.200 (28.97)	305s
Peak Cladding Temperature, ºC (ºF)	685.5 (1266.0)	10s

Table 15.5-9b

ATWS Load Rejection with a Single Failure in the Turbine Bypass System Sequence of Events

Time (s)	Event
0	Generator Load Rejection
3	ATWS trip set at high pressure
7	SRVs open
15	IC flow starts
76	Level drops below L2 set point
86	HPCRD flow starts
106	MSIV Closure starts
117	Level drops below L1 set point
188	SLCS injection starts
710	Hot Standby Volume of Boron injected into Bypass



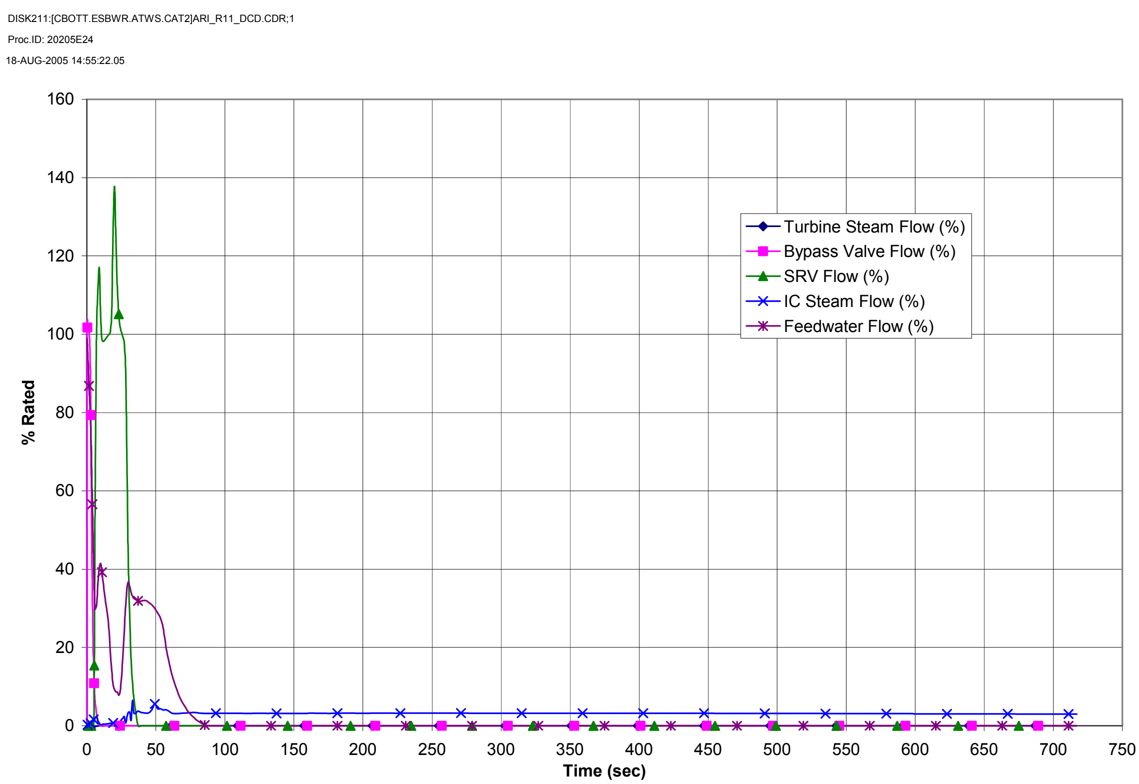
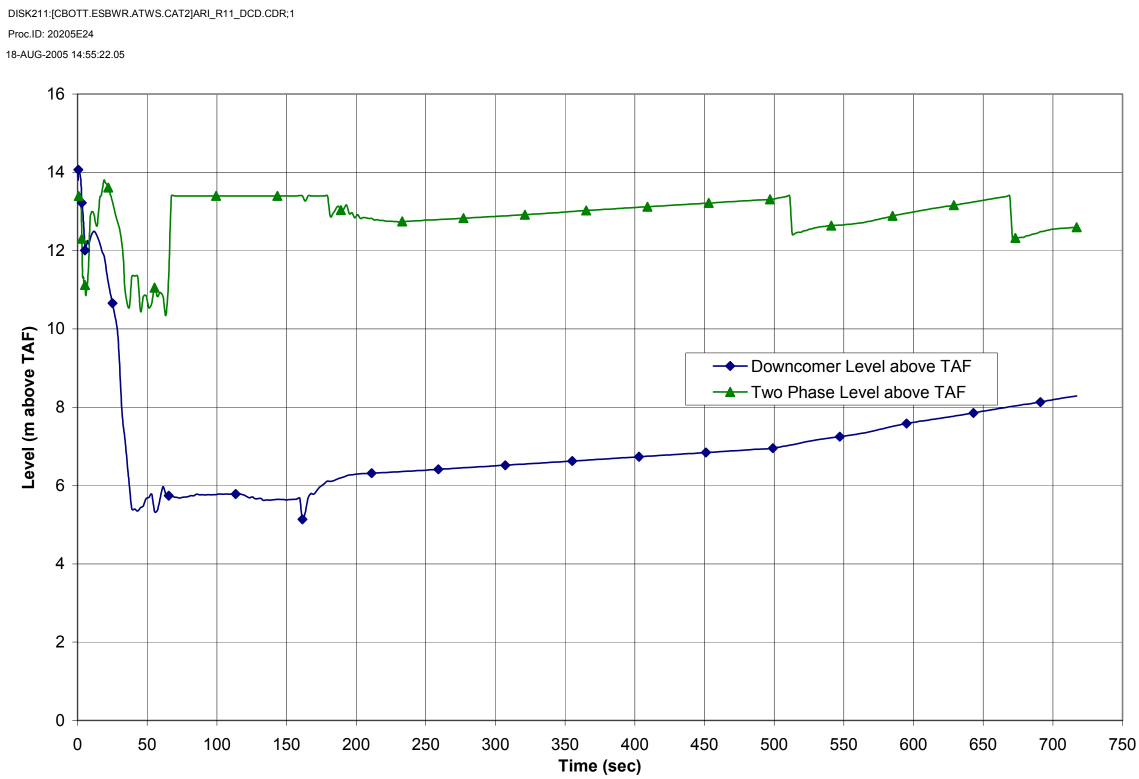


Figure 15.5-1a. MSIV Closure with ARI



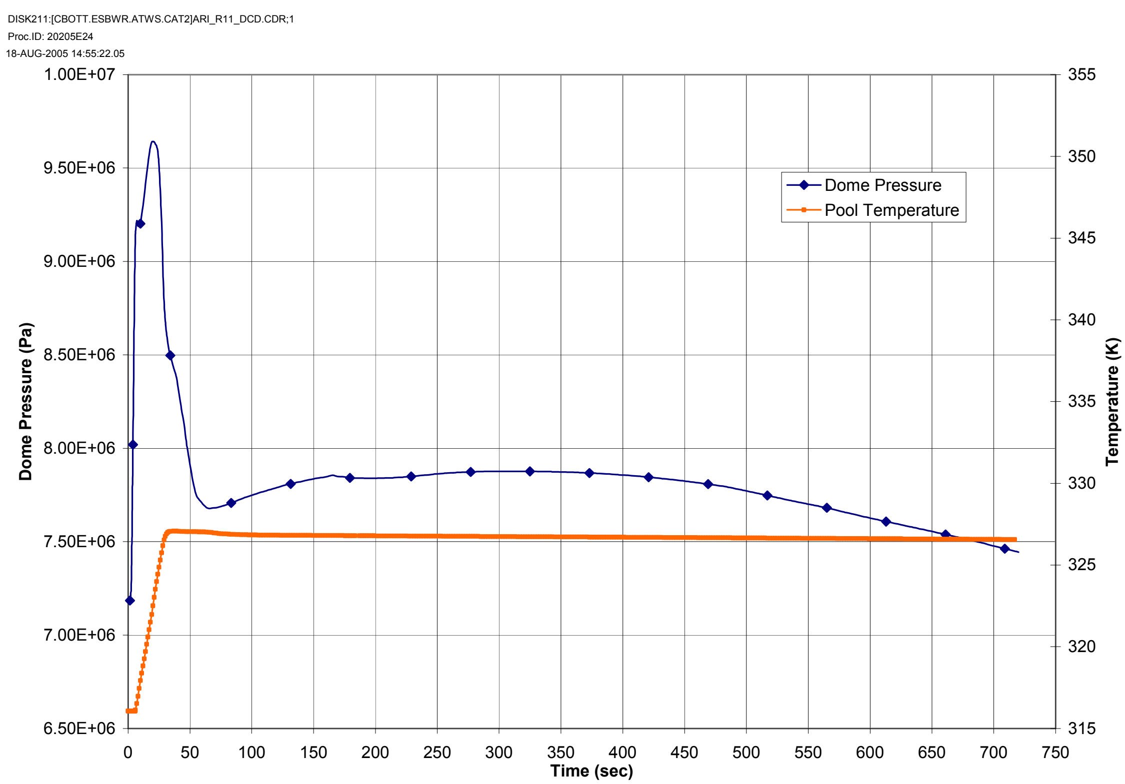
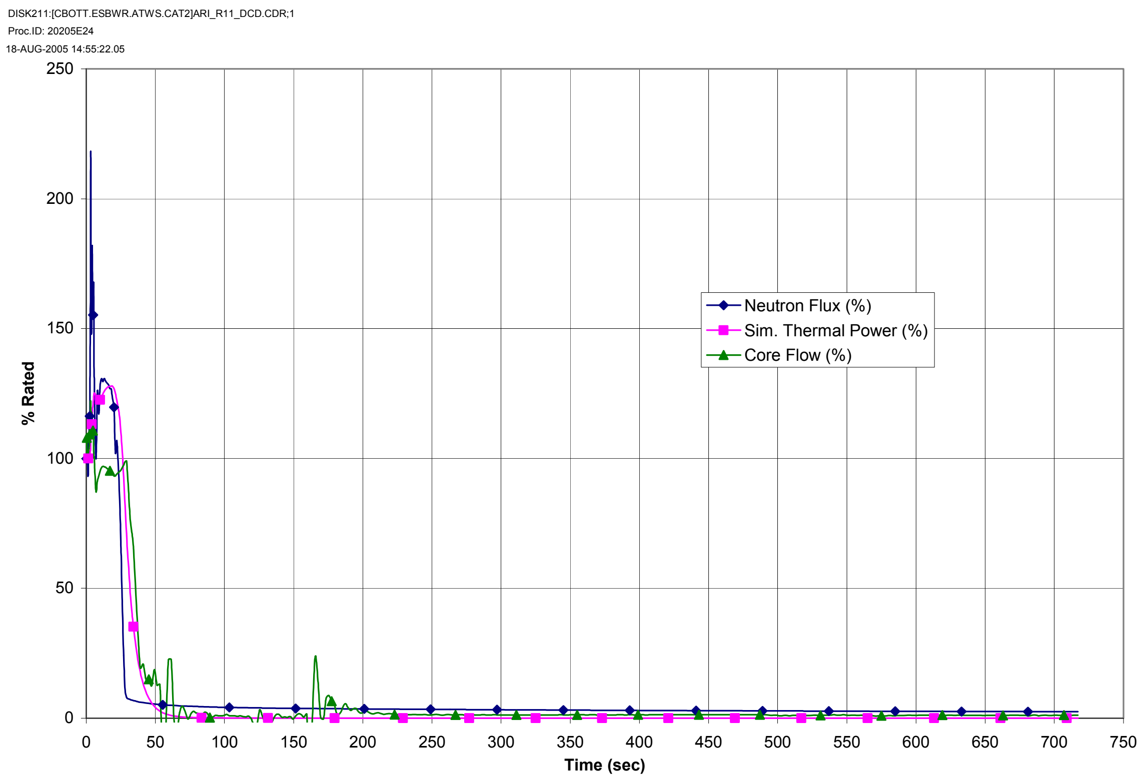


Figure 15.5-1b. MSIV Closure with ARI



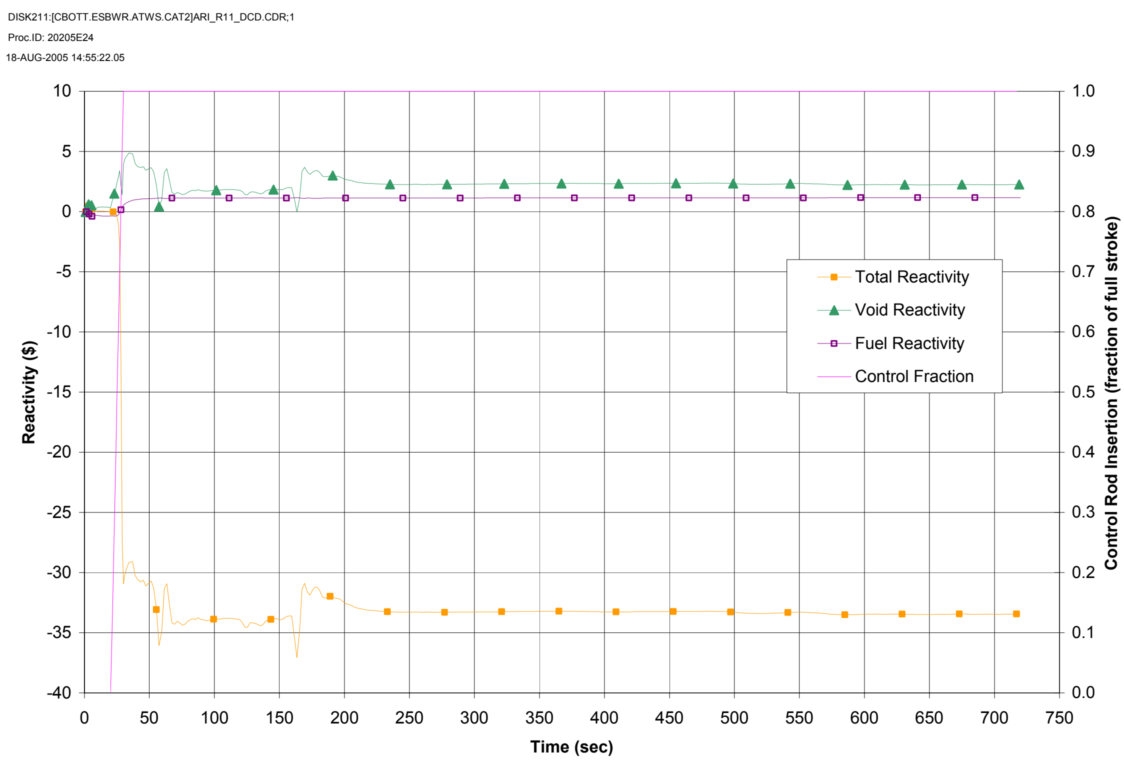


Figure 15.5-1c. MSIV Closure with ARI

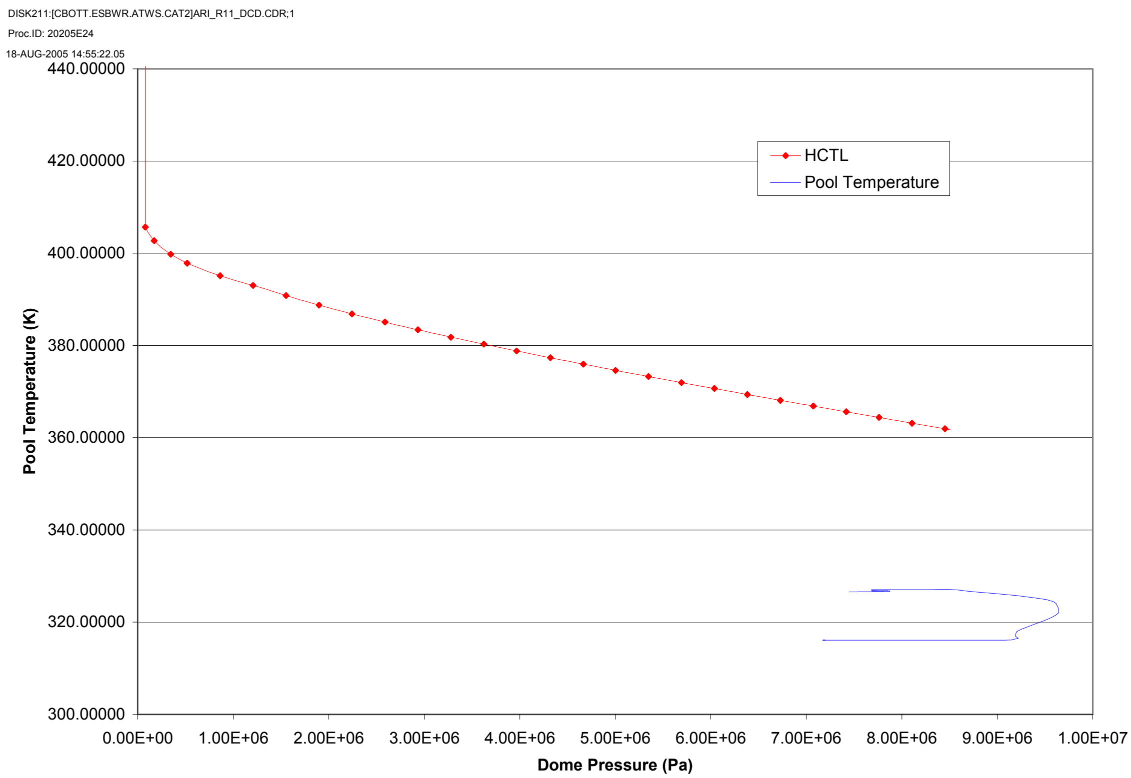
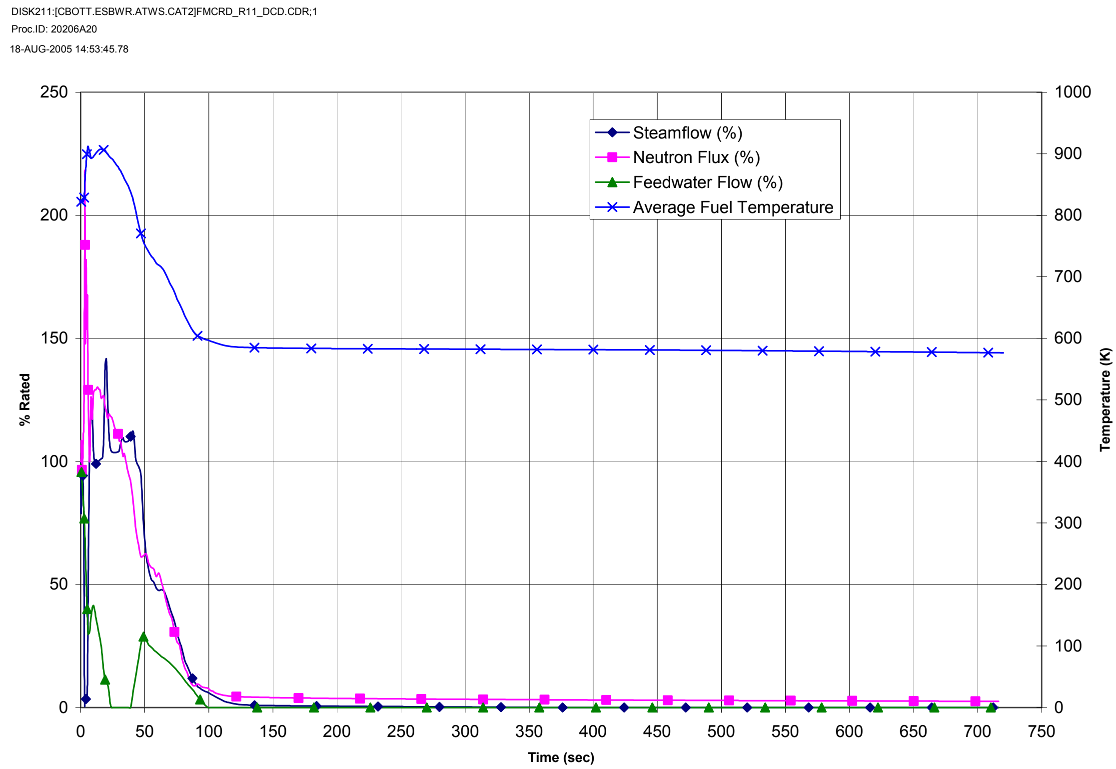


Figure 15.5-1d. MSIV Closure with ARI



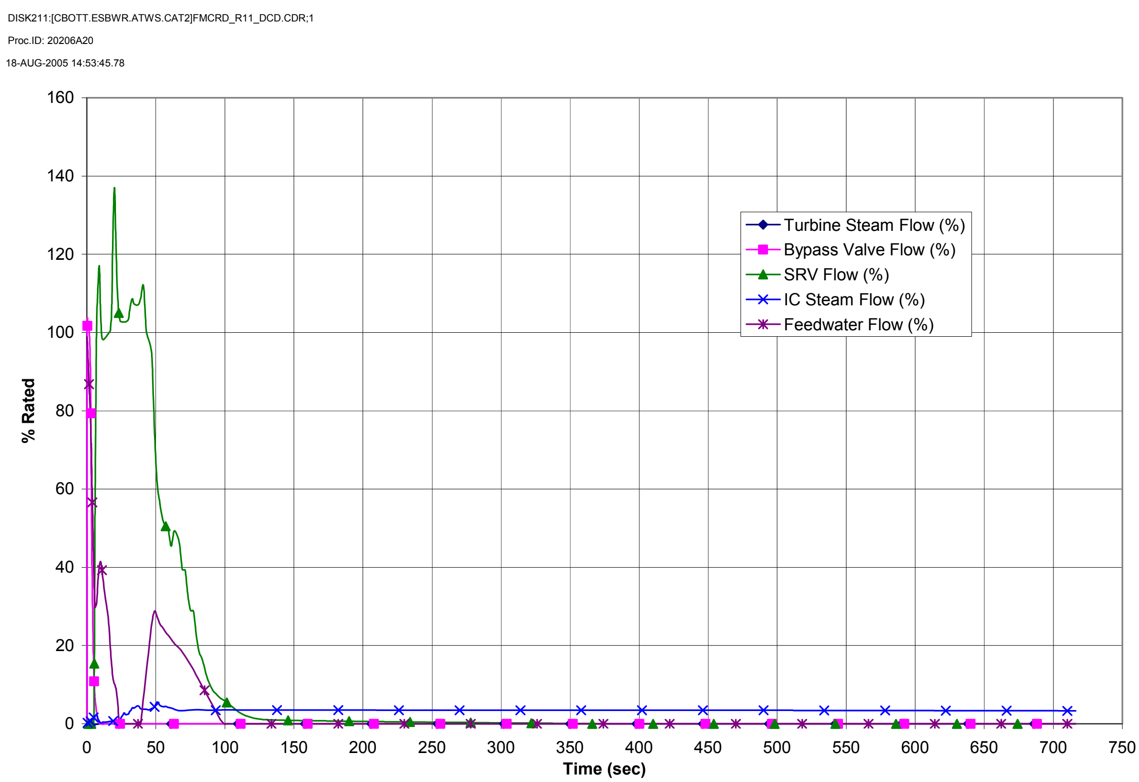


Figure 15.5-2a. MSIV Closure with FMCRD Run-in

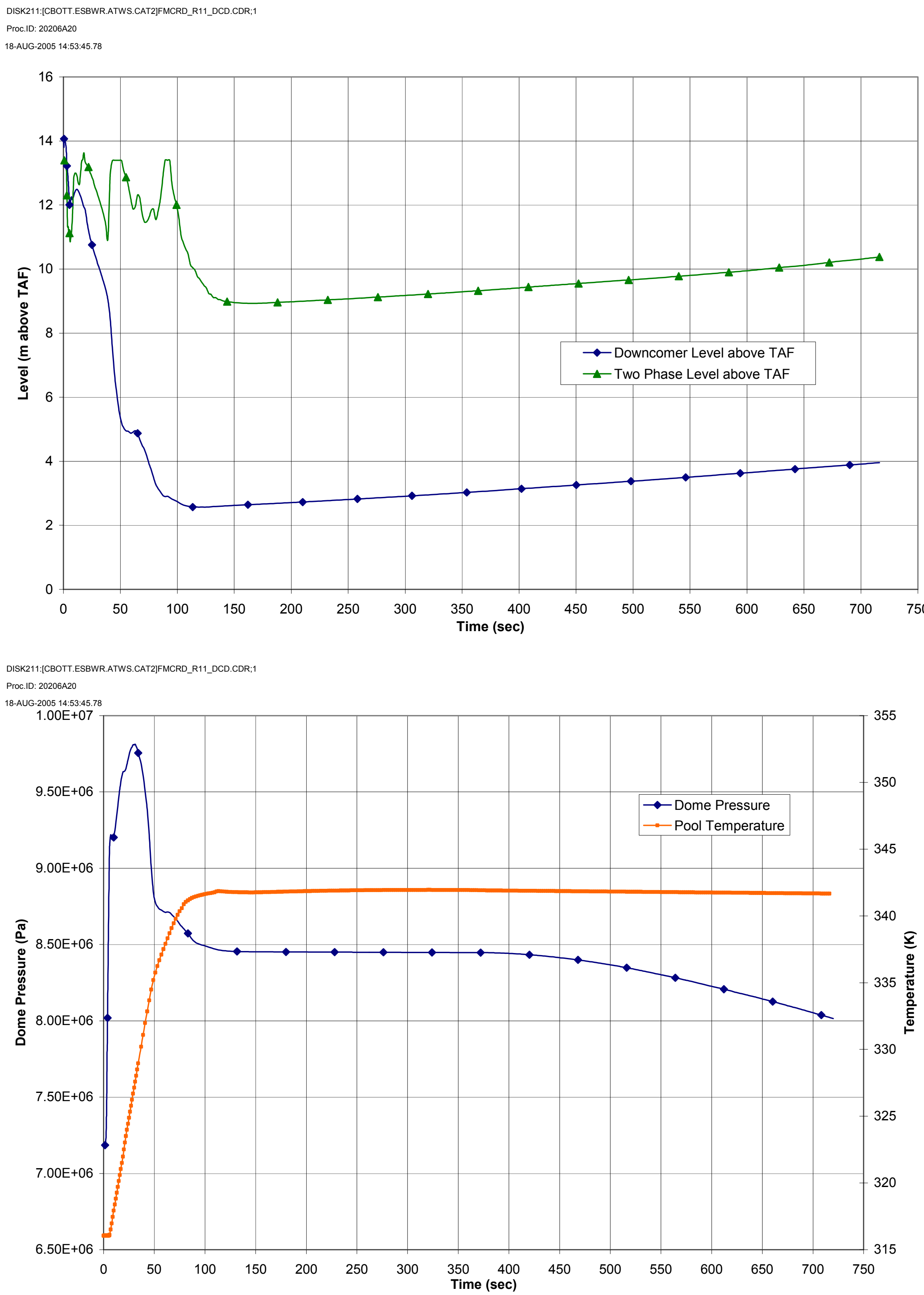
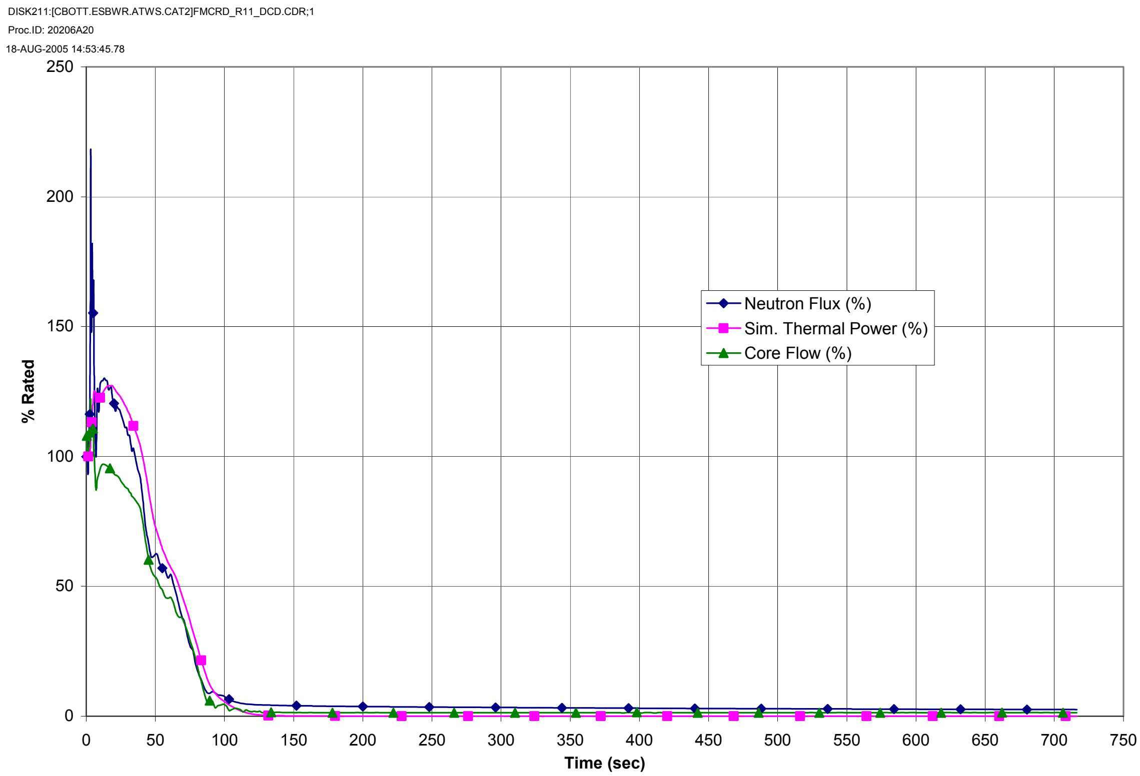


Figure 15.5-2b. MSIV Closure with FMCRD Run-in



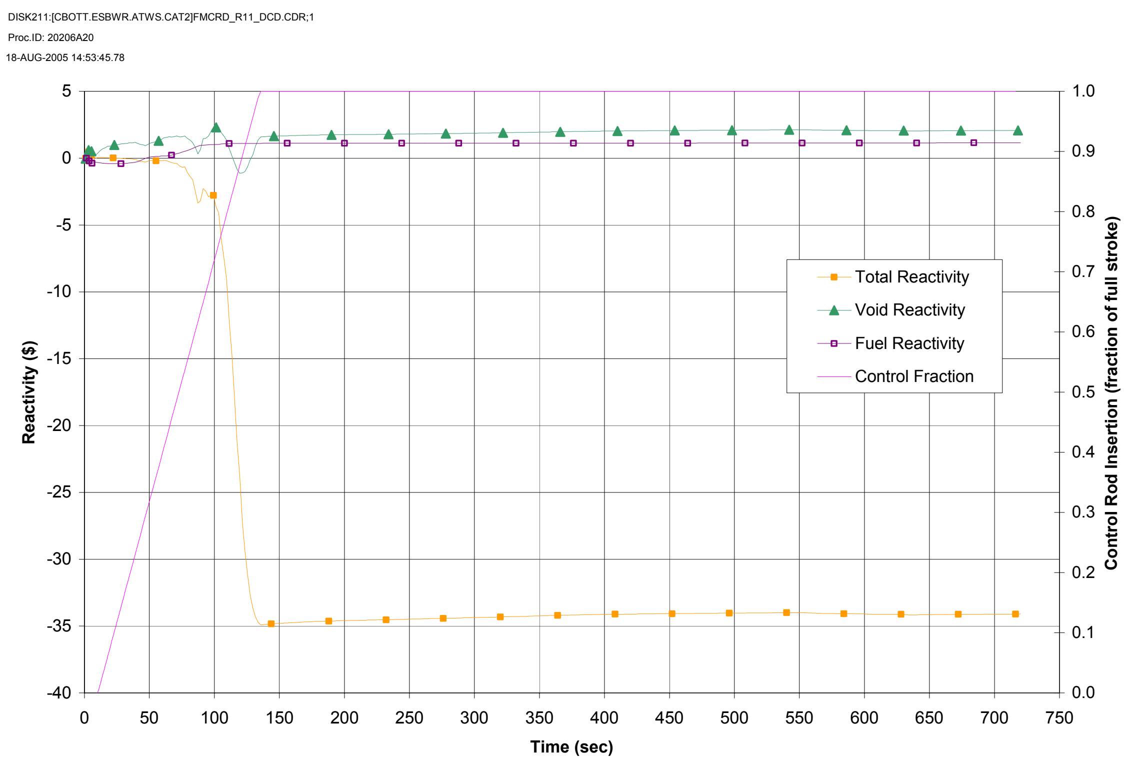


Figure 15.5-2c. MSIV Closure with FMCRD Run-in

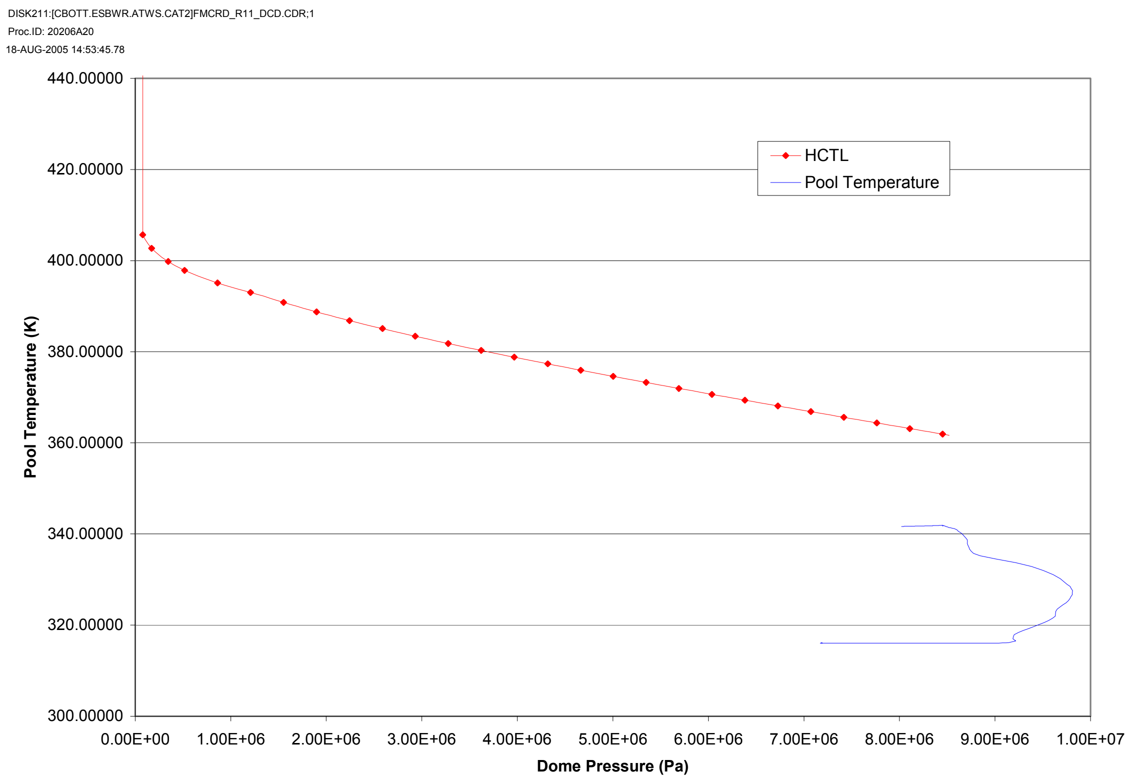
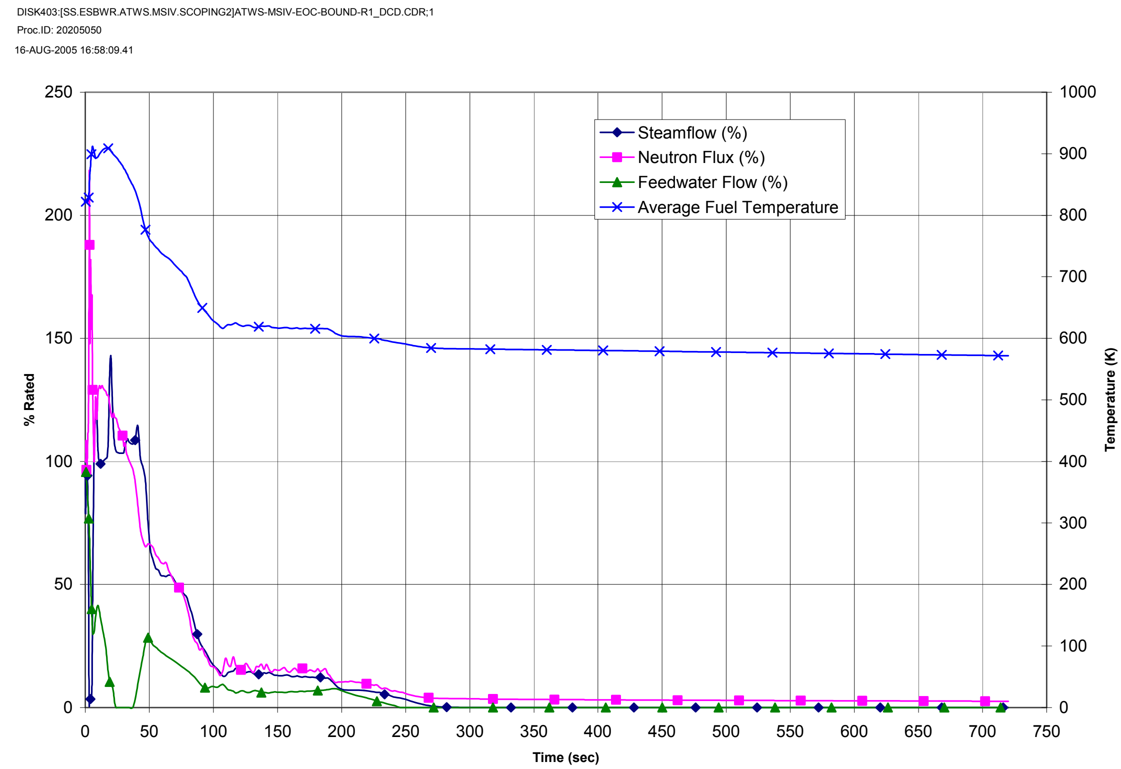


Figure 15.5-2d. MSIV Closure with FMCRD Run-in



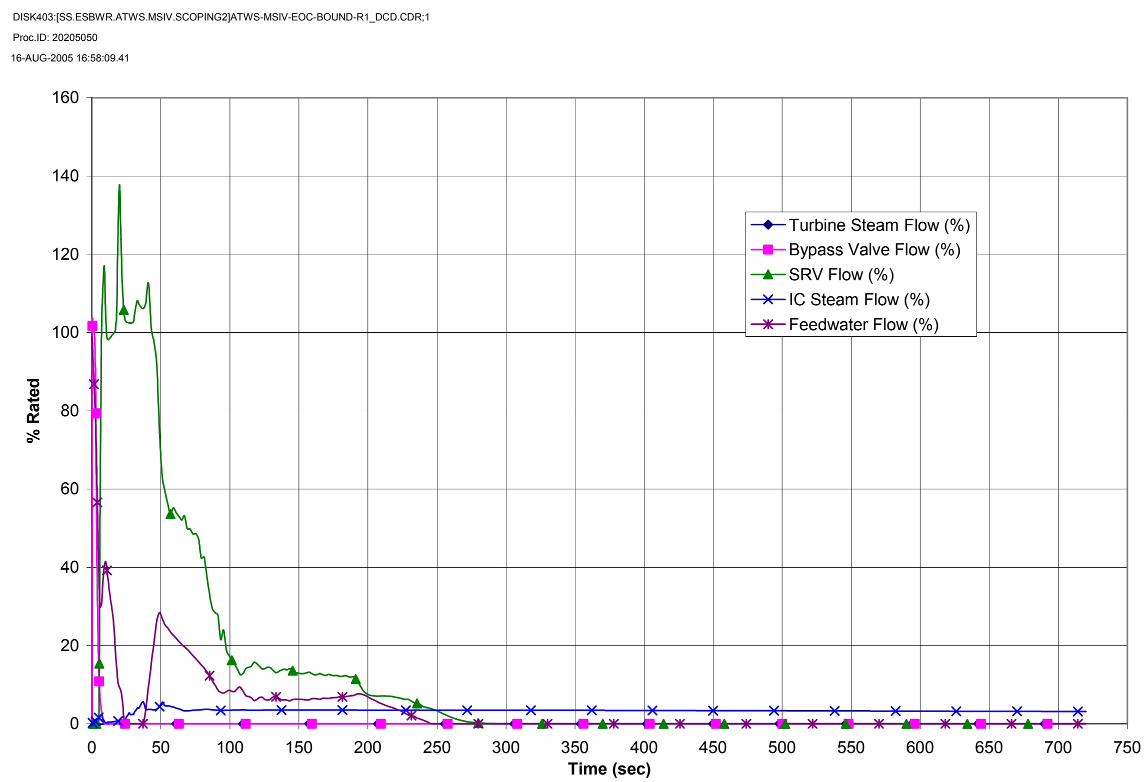
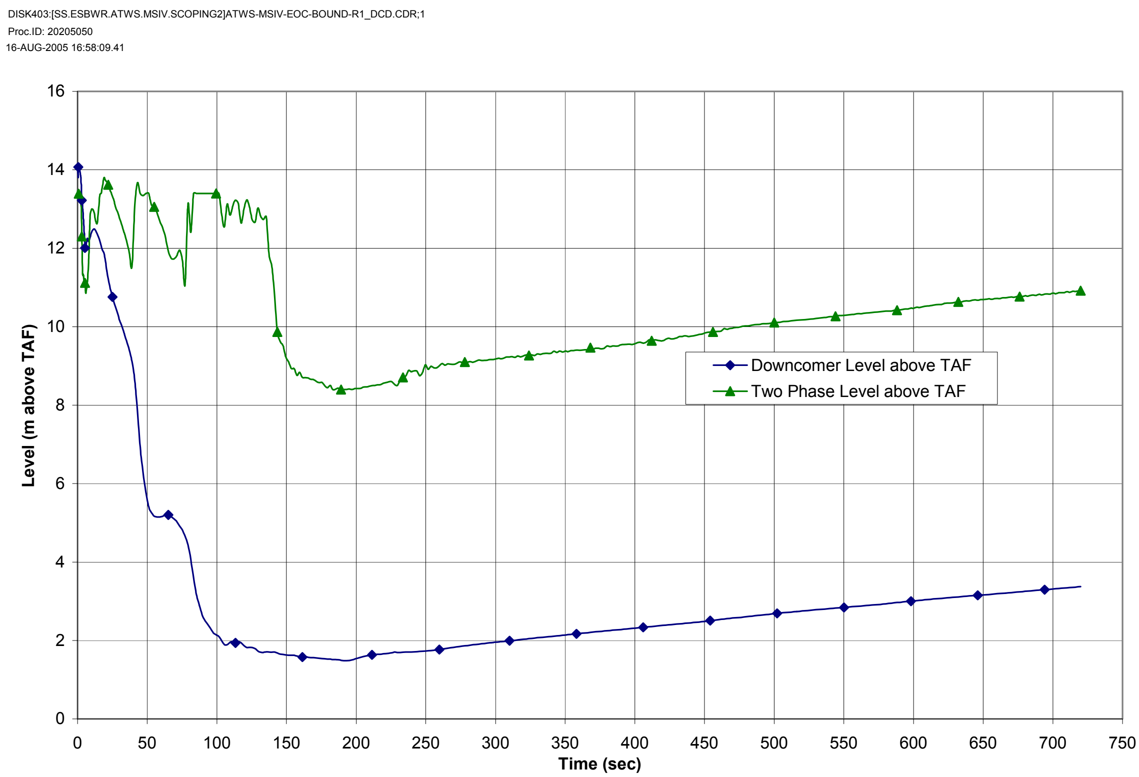


Figure 15.5-3a. MSIV Closure with Boron Injection



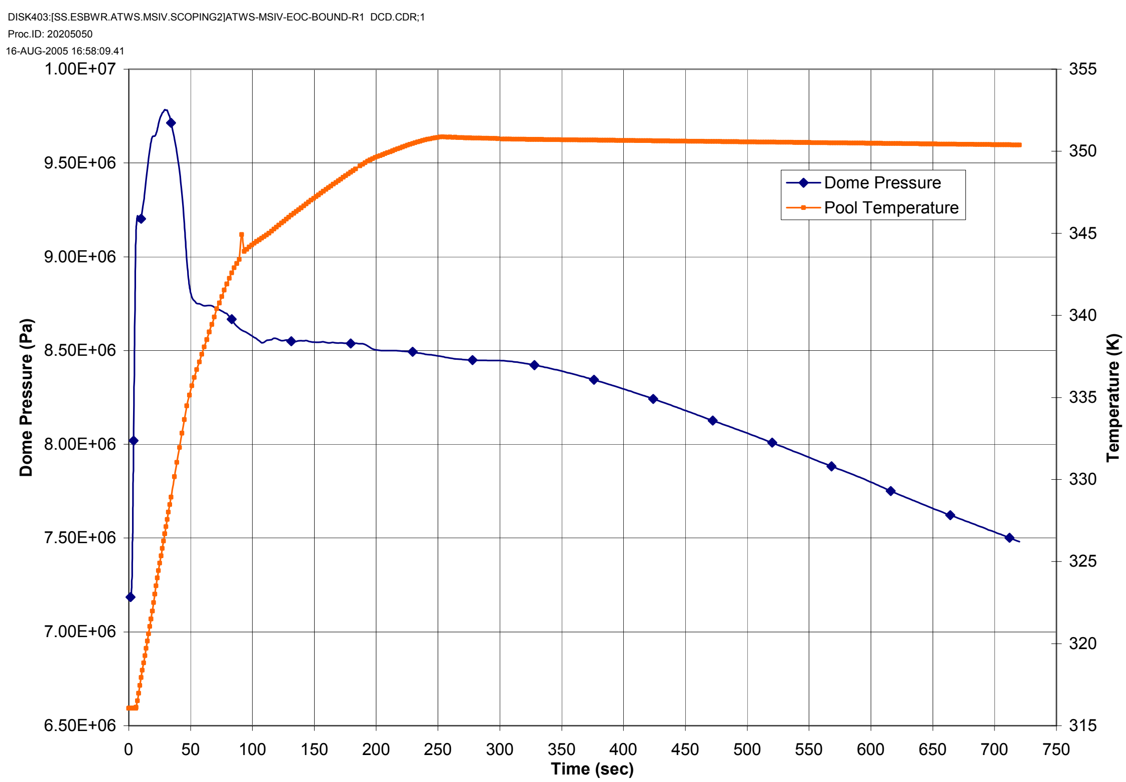
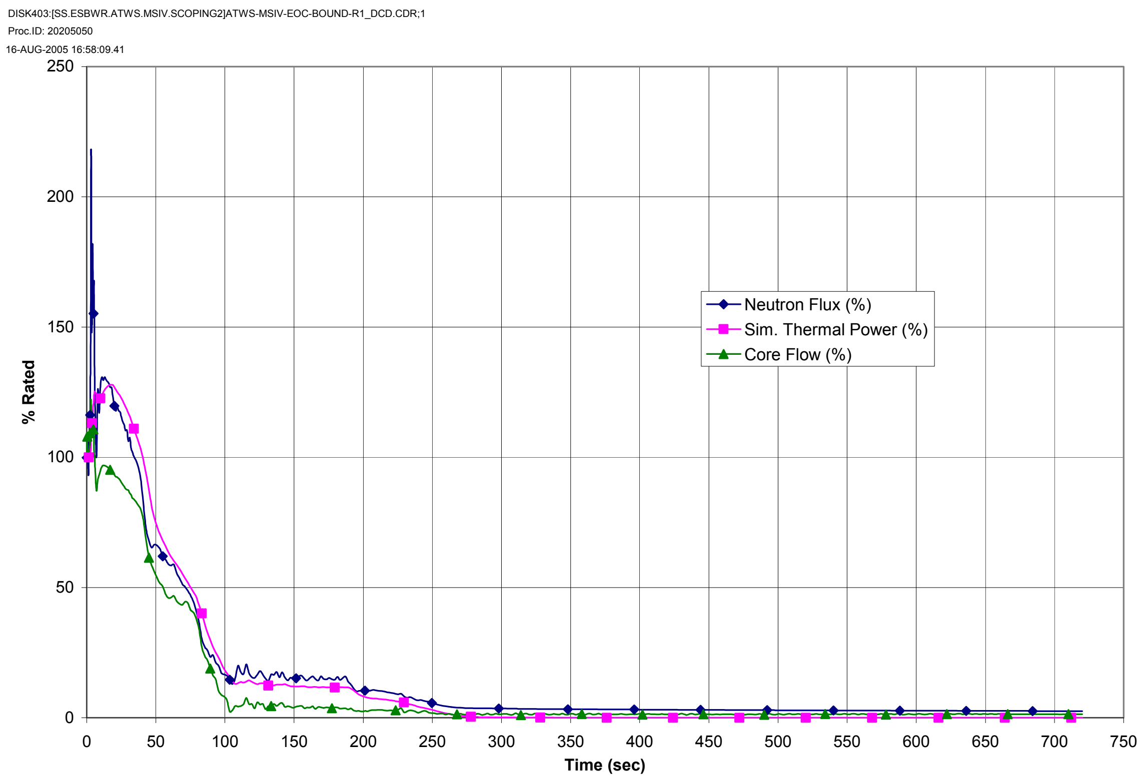


Figure 15.5-3b. MSIV Closure with Boron Injection



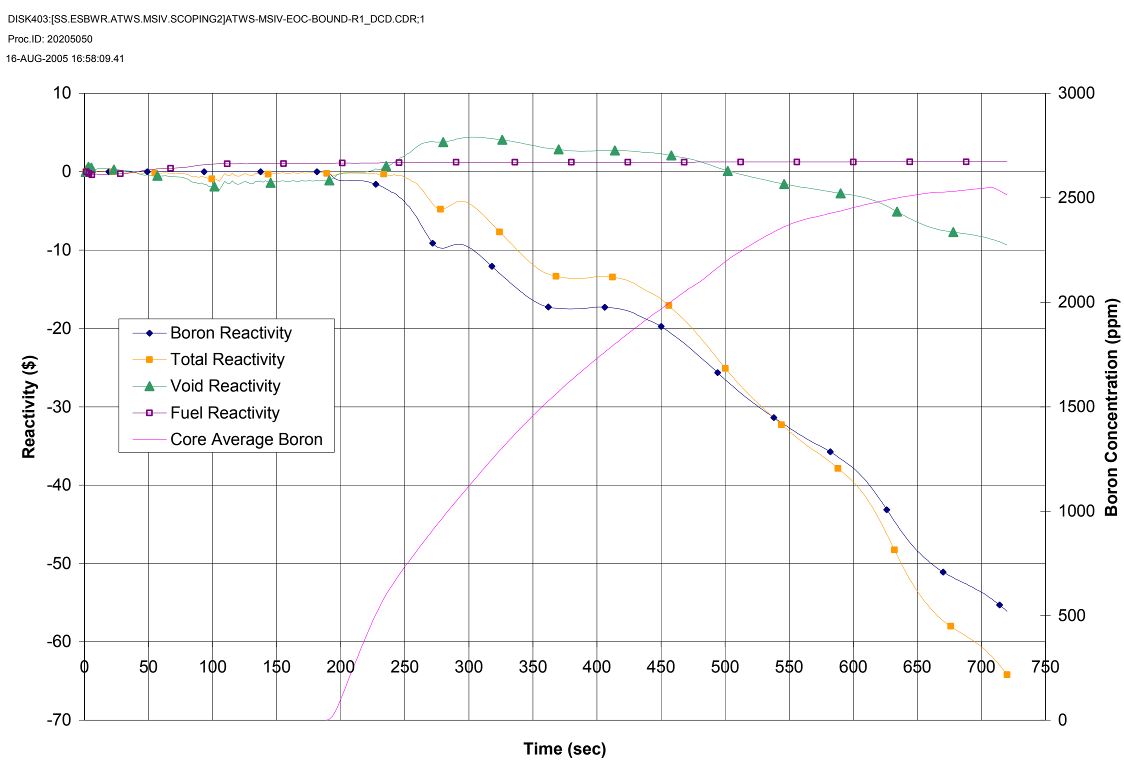
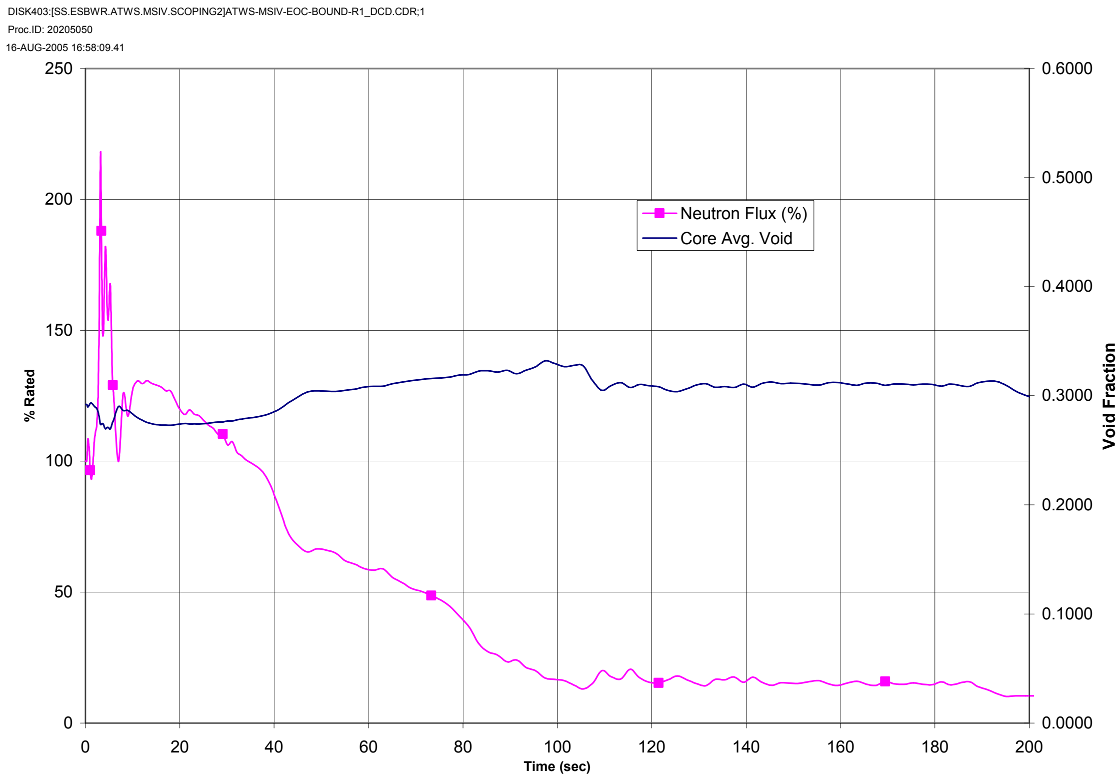


Figure 15.5-3c. MSIV Closure with Boron Injection



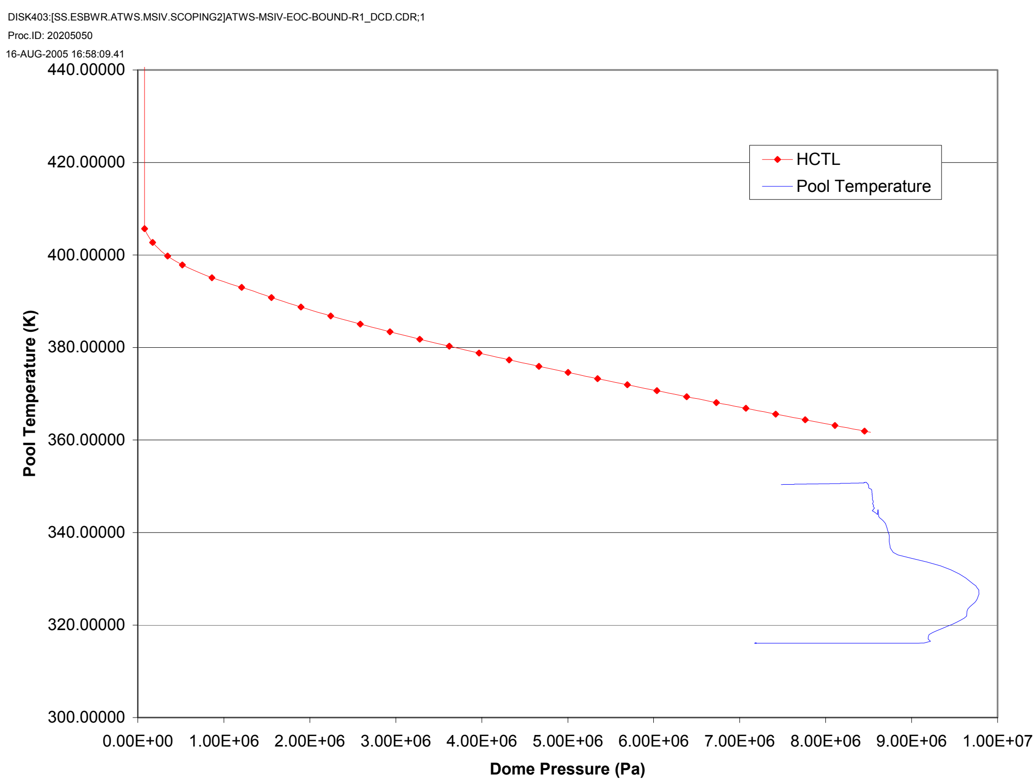
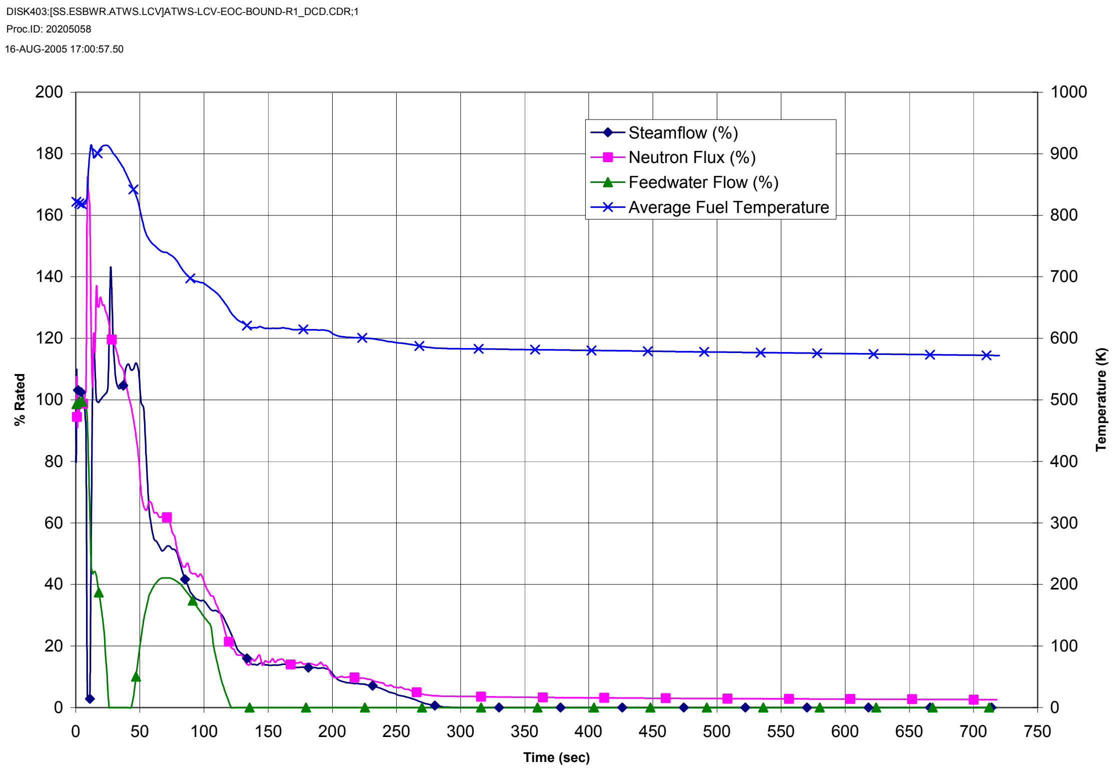


Figure 15.5-3d. MSIV Closure with Boron Injection



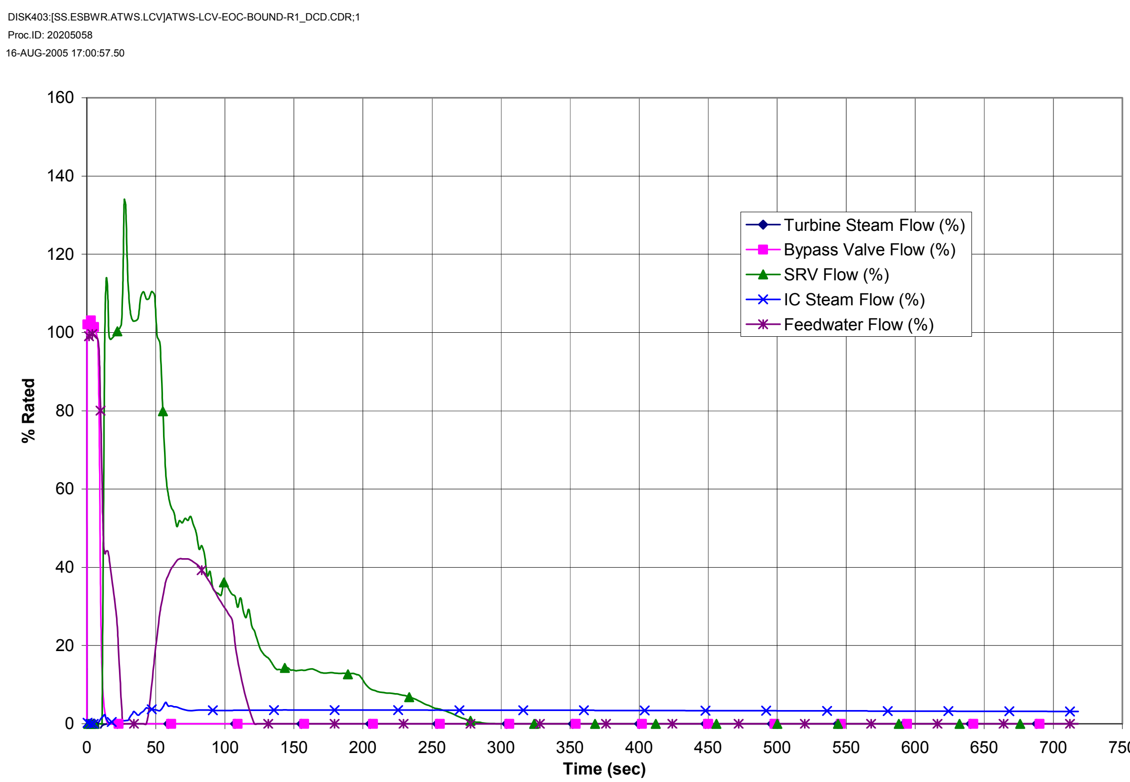
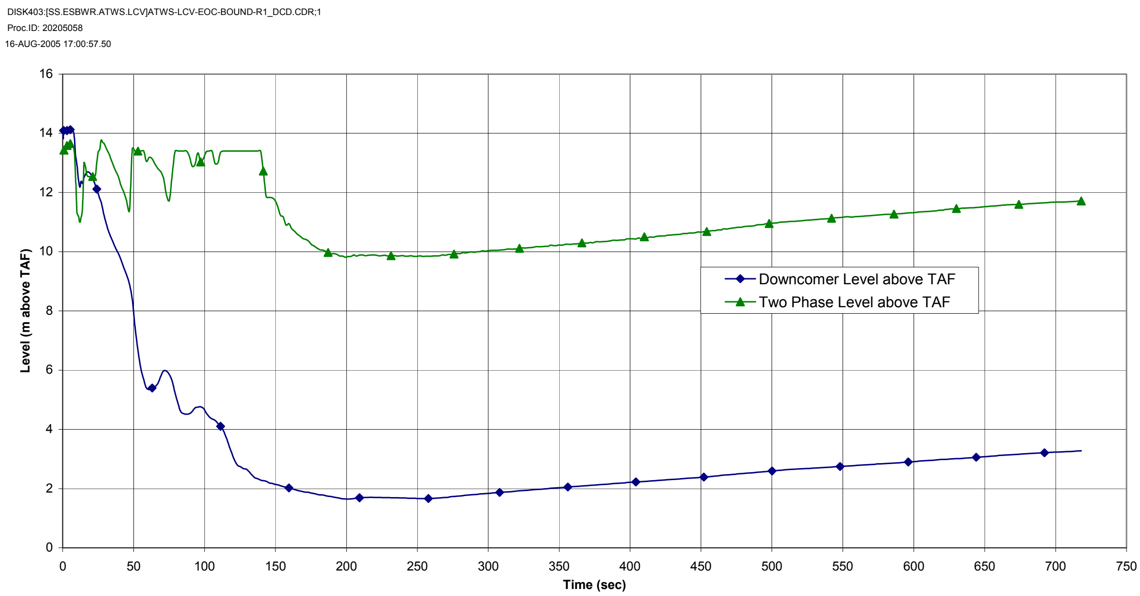


Figure 15.5-4a. Loss of Condenser Vacuum with Boron Injection



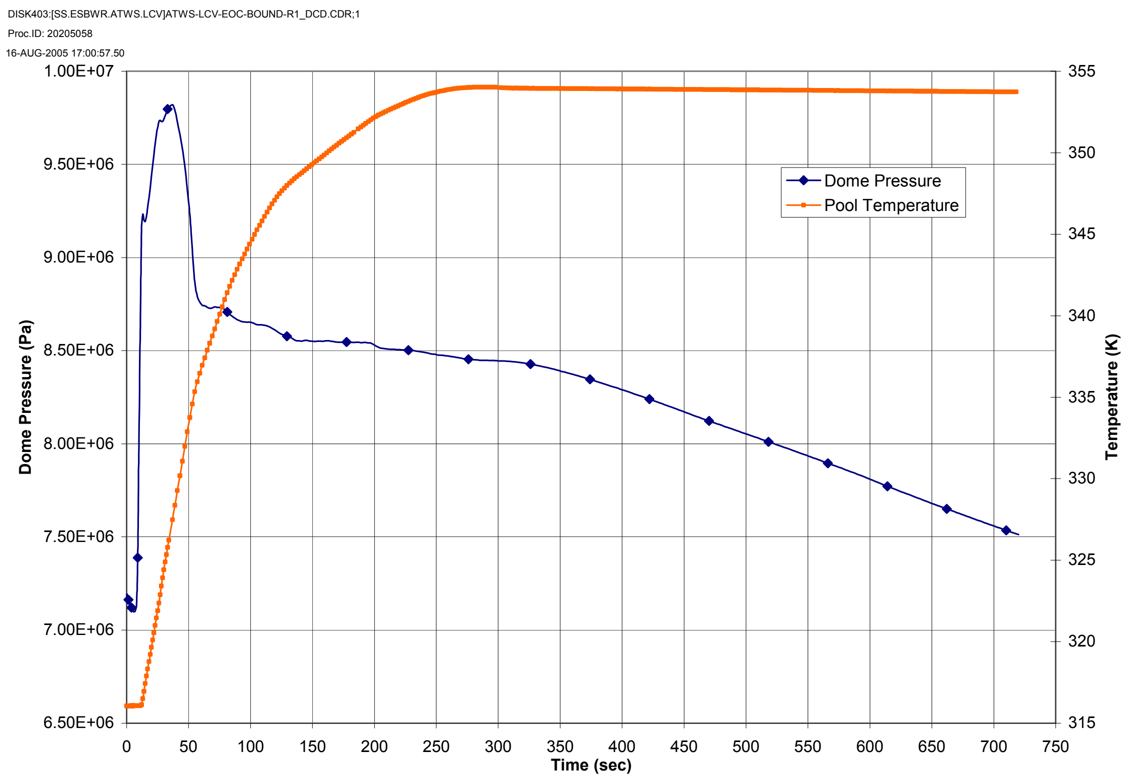
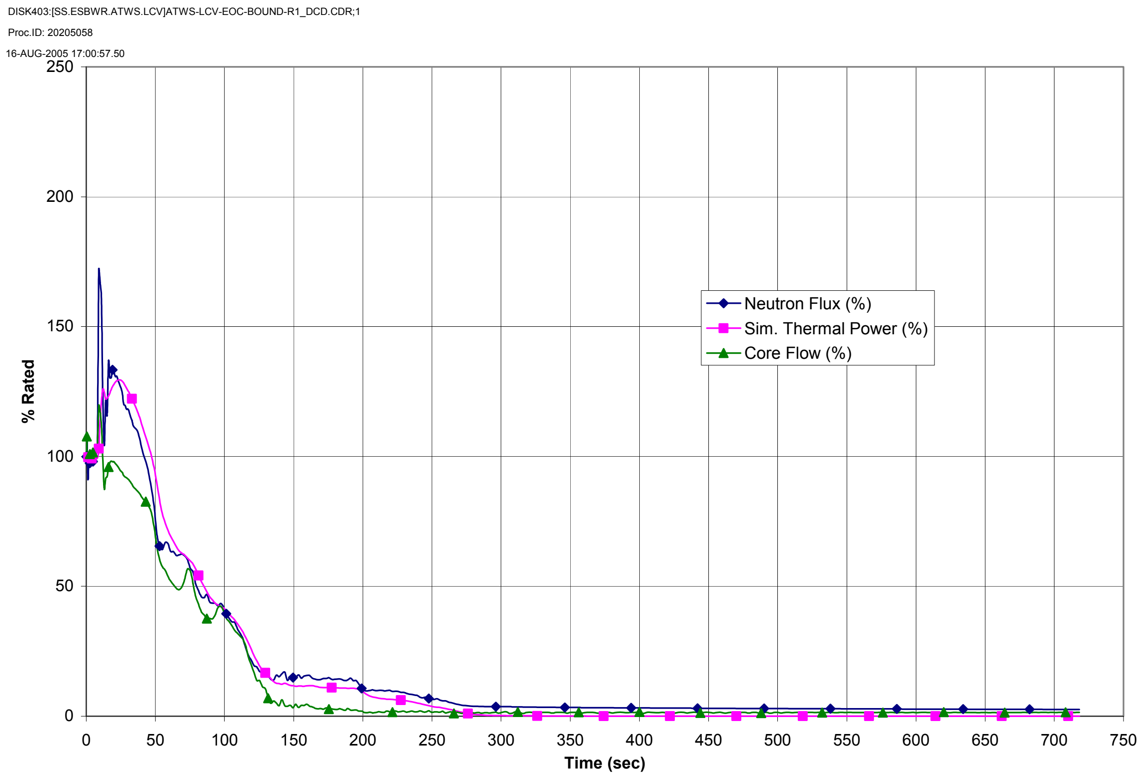


Figure 15.5-4b. Loss of Condenser Vacuum with Boron Injection



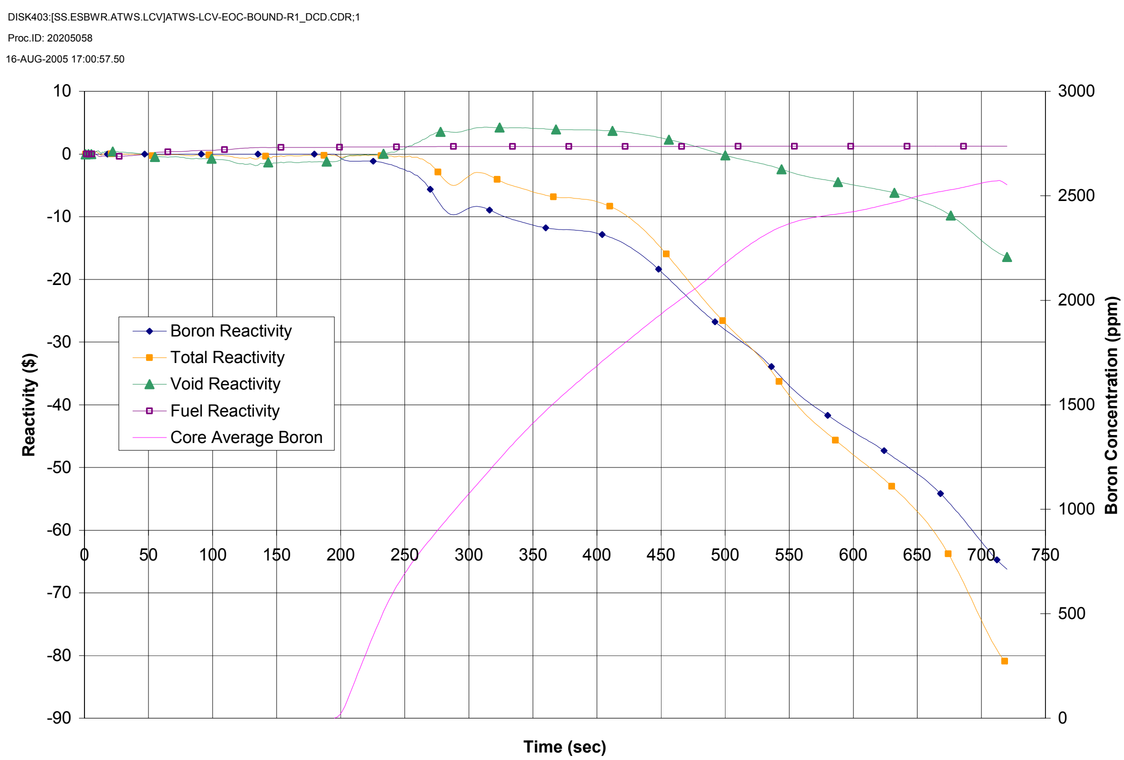


Figure 15.5-4c. Loss of Condenser Vacuum with Boron Injection

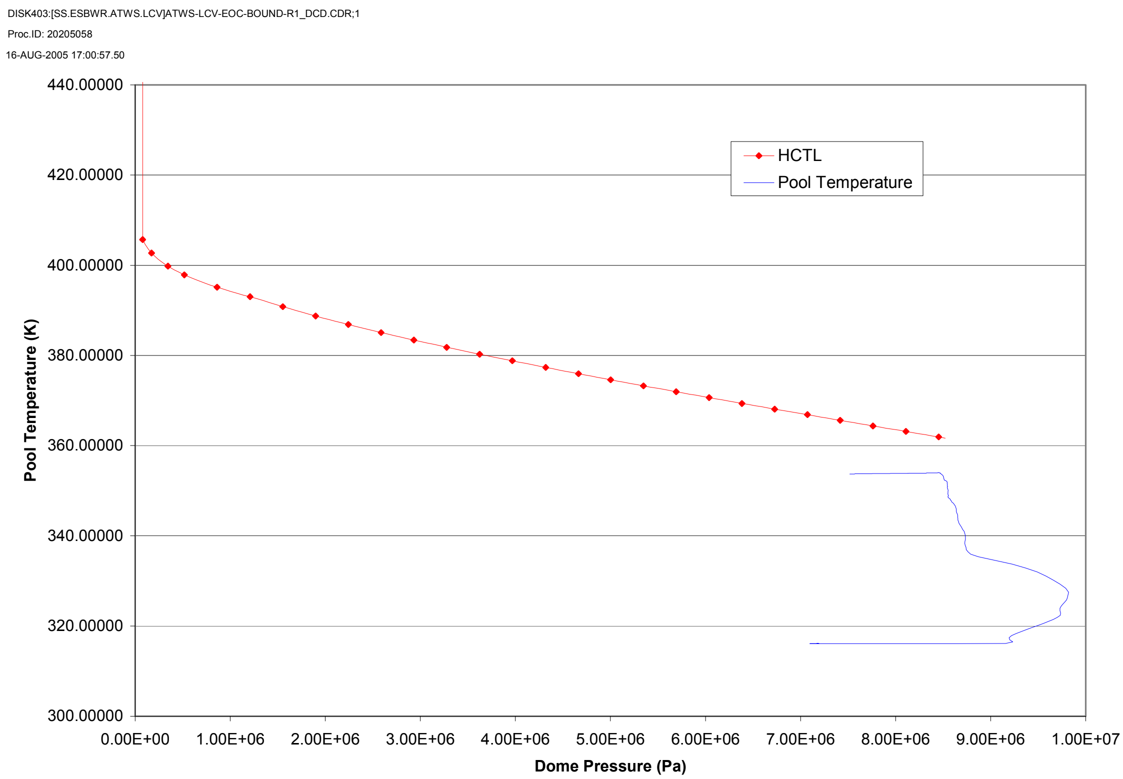
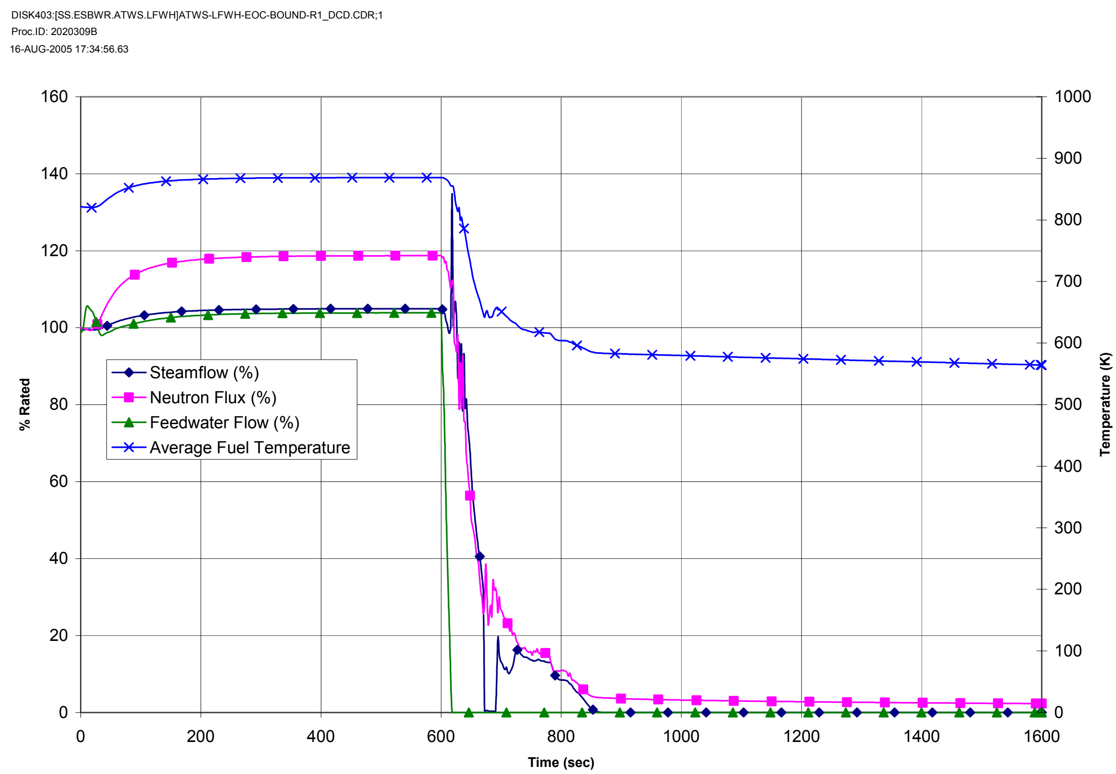


Figure 15.5-4d. Loss of Condenser Vacuum with Boron Injection



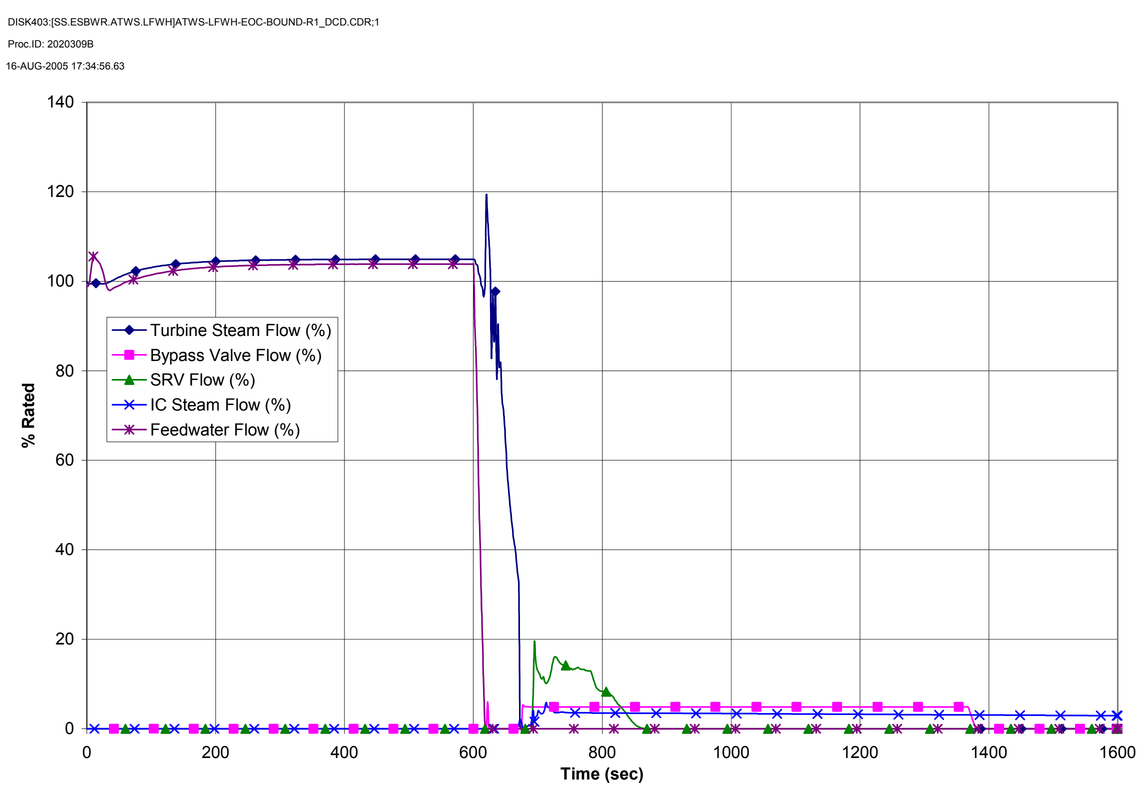


Figure 15.5-5a. Loss of Feedwater Heating with Boron Injection

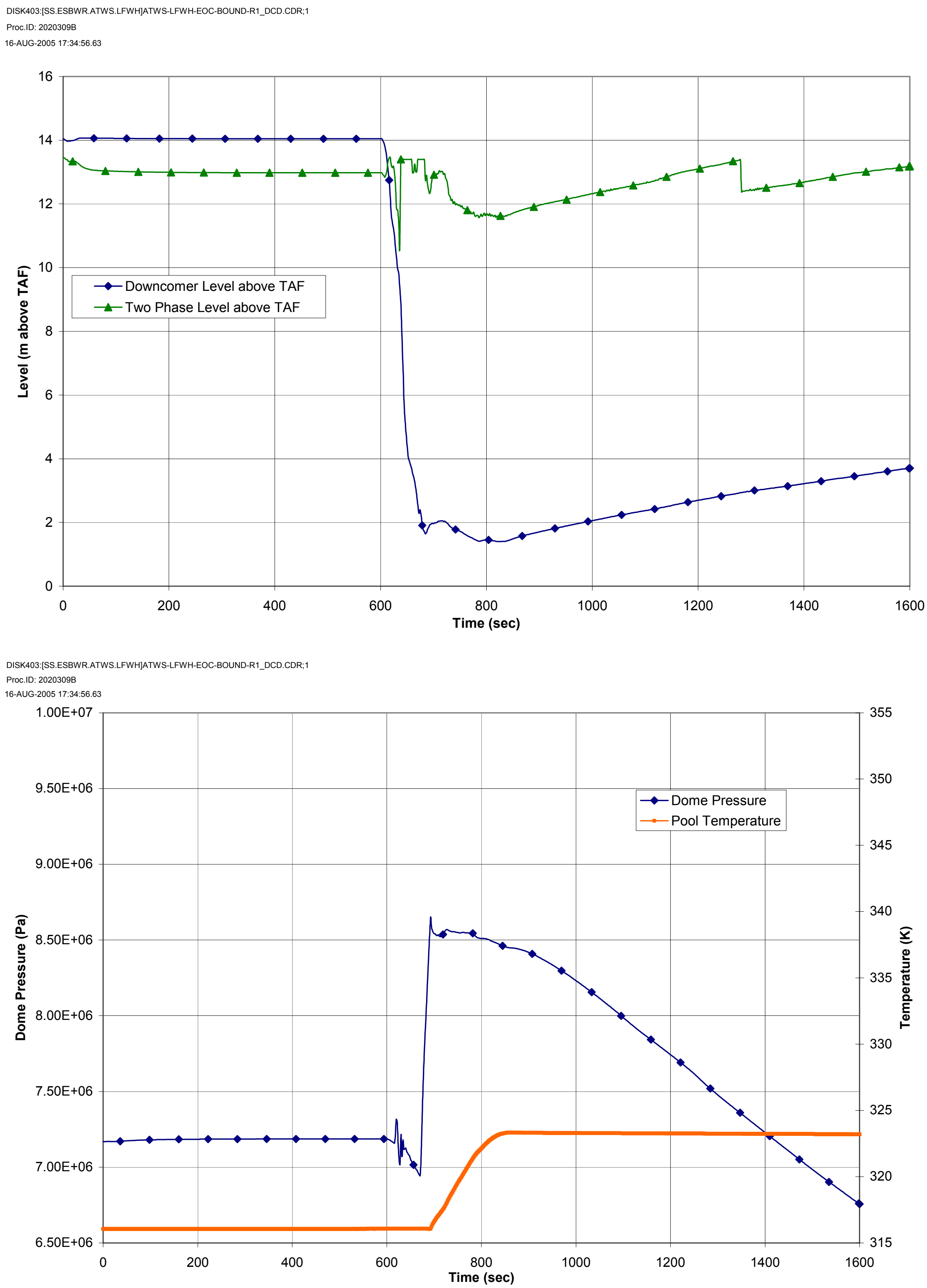
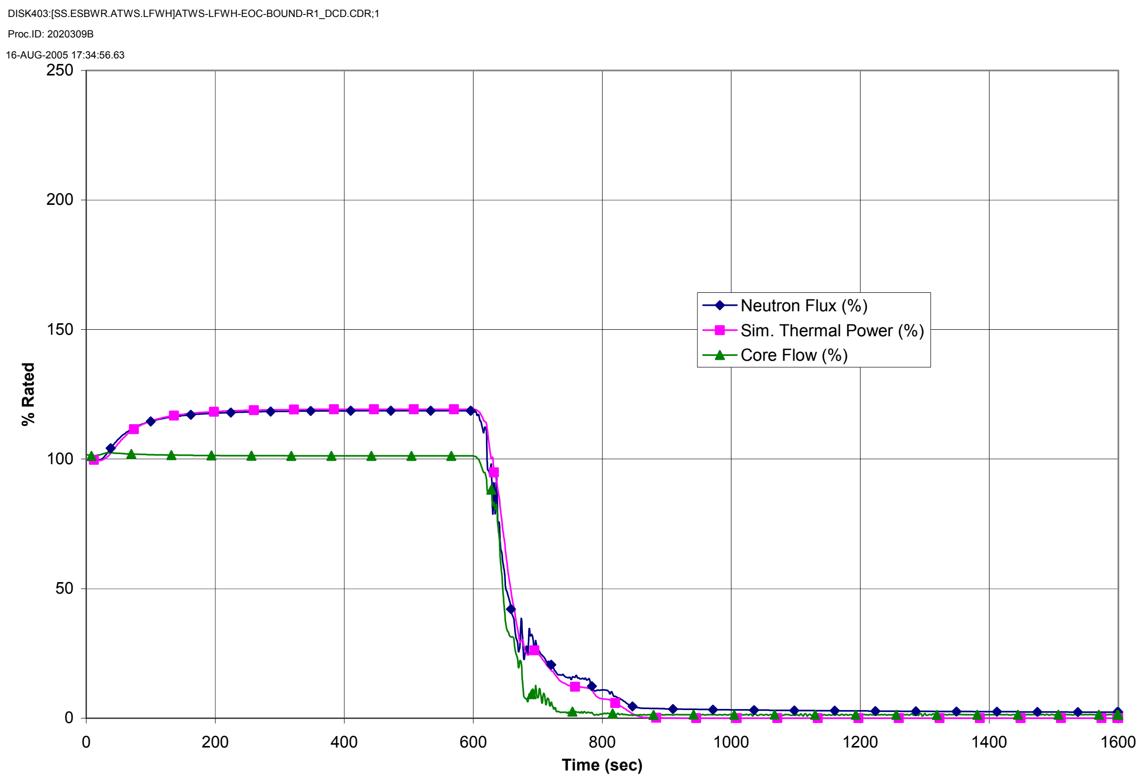


Figure 15.5-5b. Loss of Feedwater Heating with Boron Injection



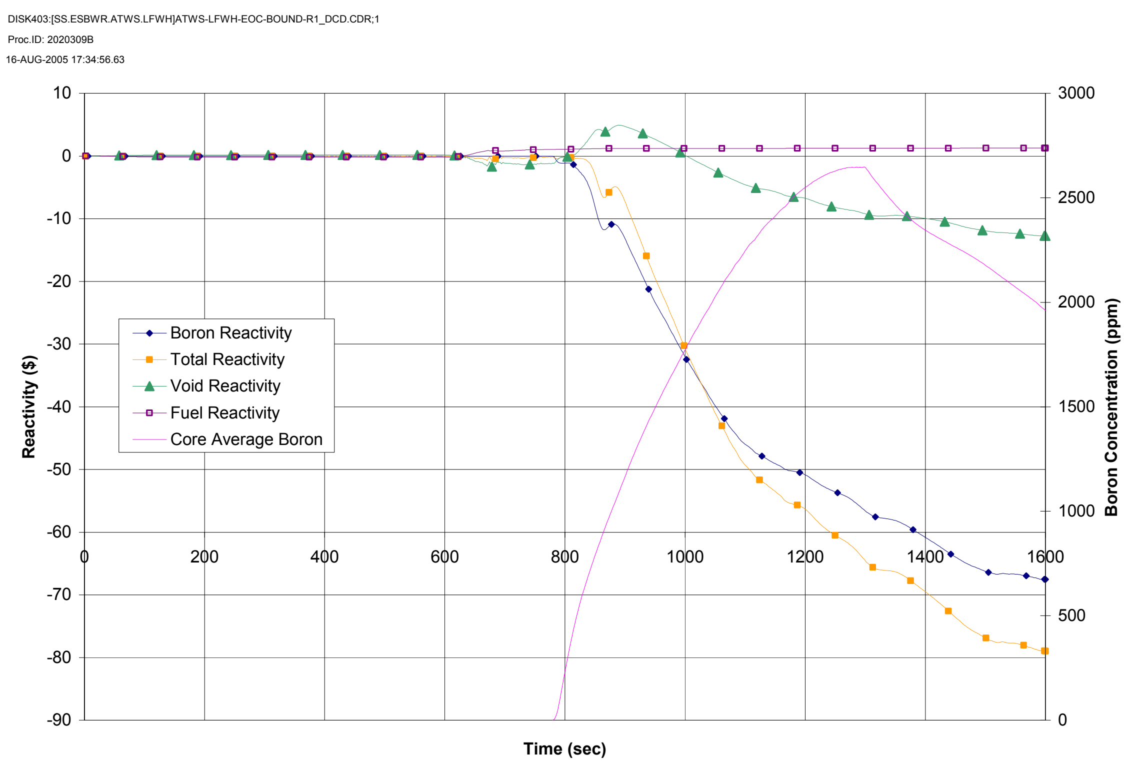


Figure 15.5-5c. Loss of Feedwater Heating with Boron Injection

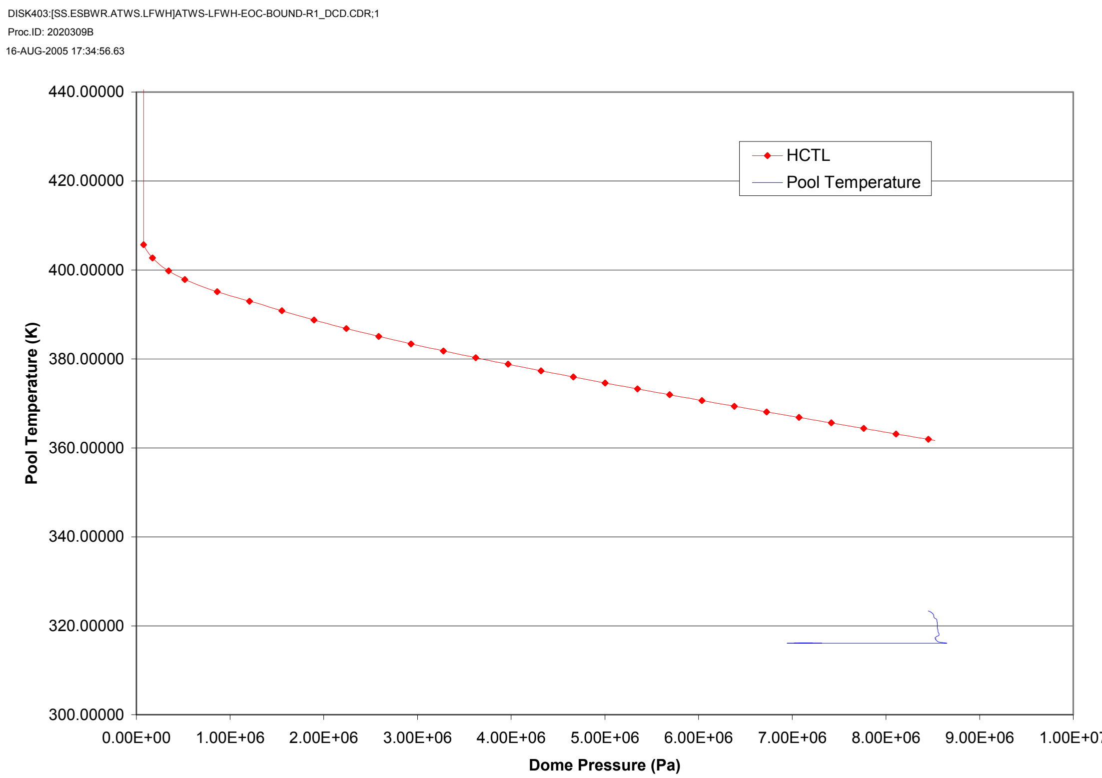
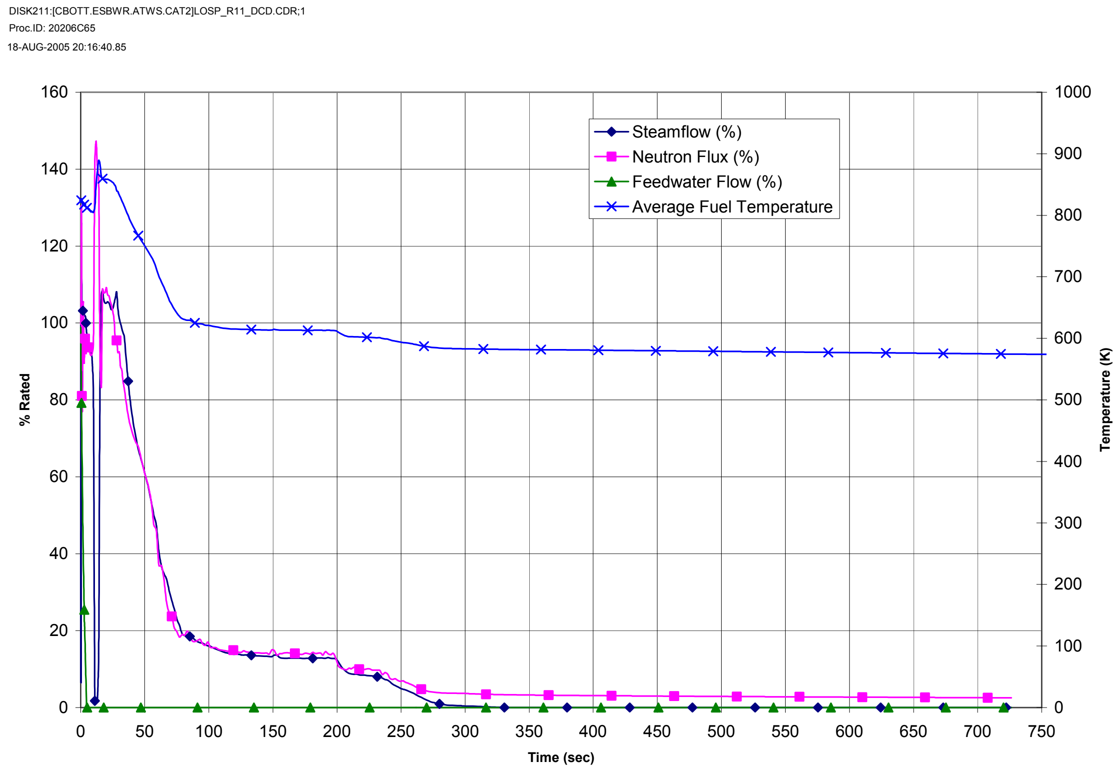


Figure 15.5-5d. Loss of Feedwater Heating with Boron Injection



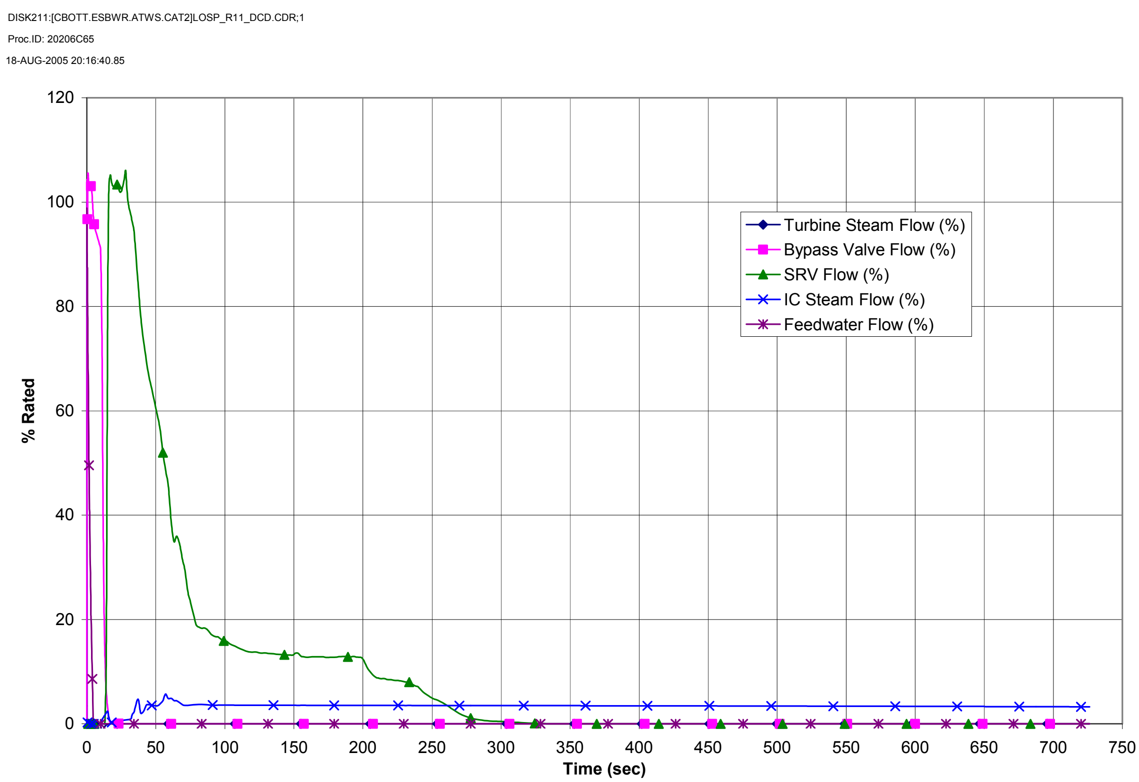


Figure 15.5-6a. Loss of Normal AC Power to Station Auxiliaries with Boron Injection

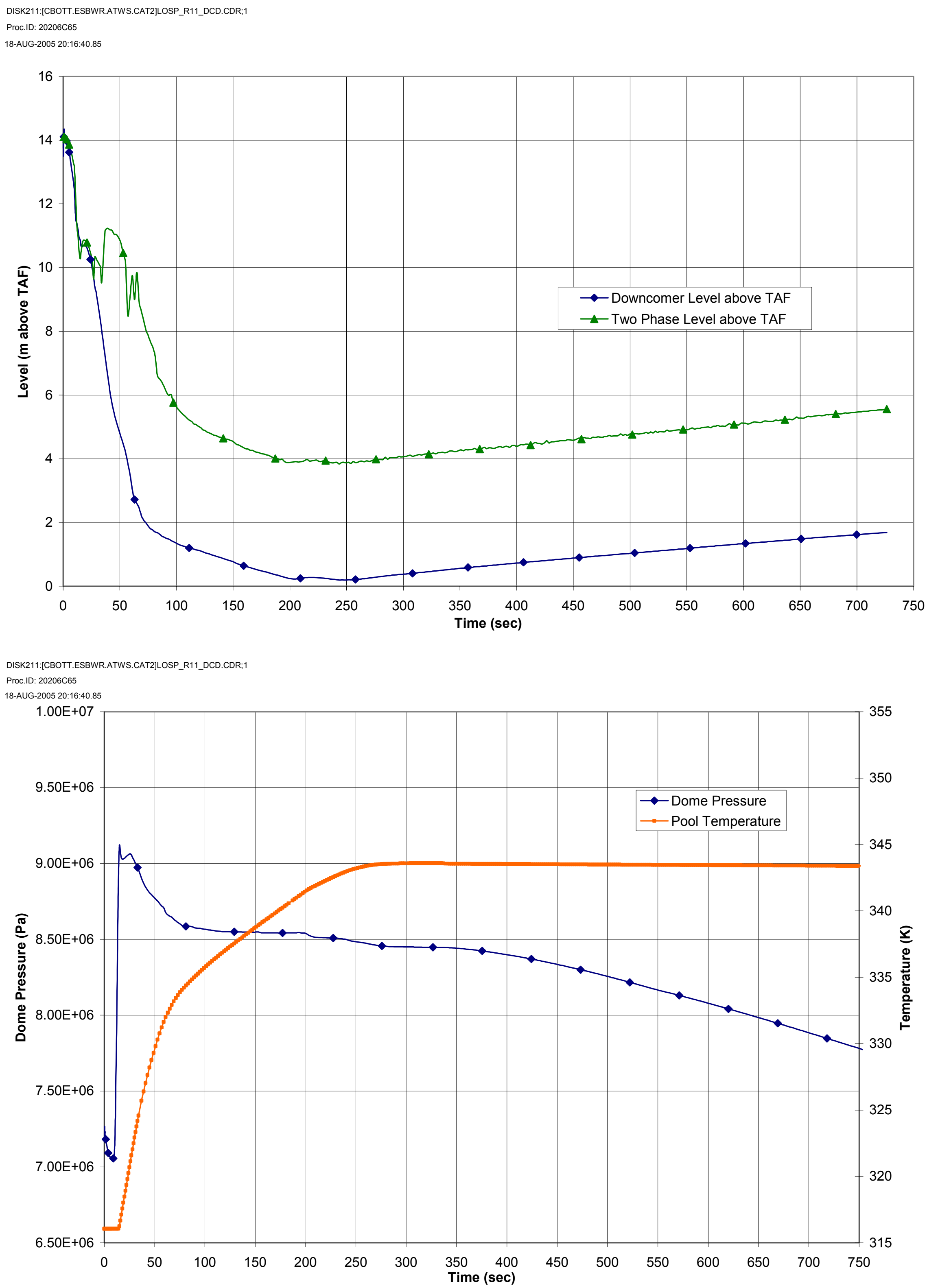
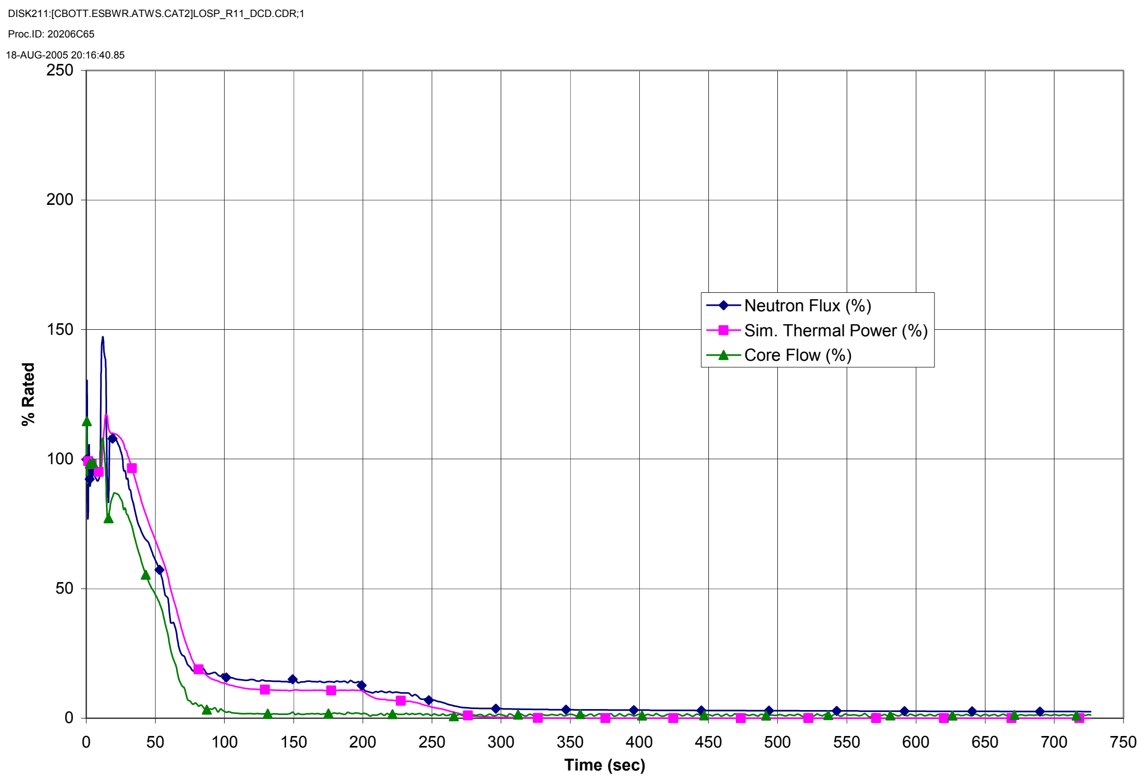


Figure 15.5-6b. Loss of Normal AC Power to Station Auxiliaries with Boron Injection



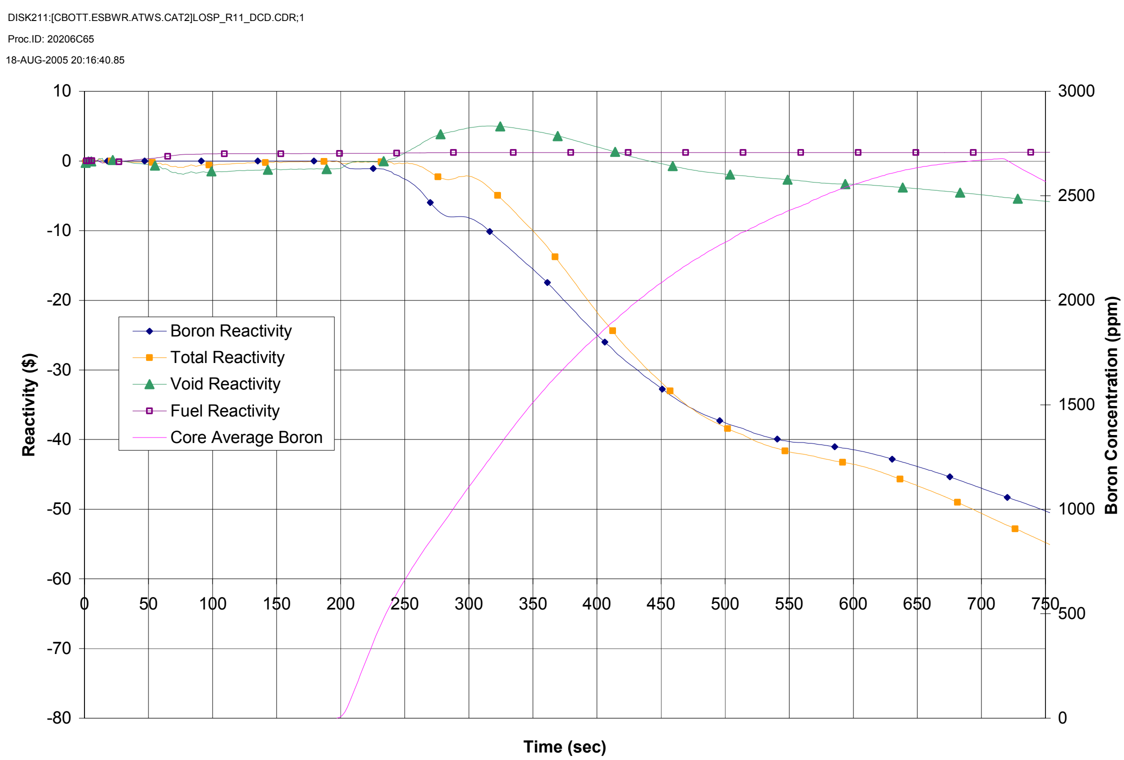


Figure 15.5-6c. Loss of Normal AC Power to Station Auxiliaries with Boron Injection

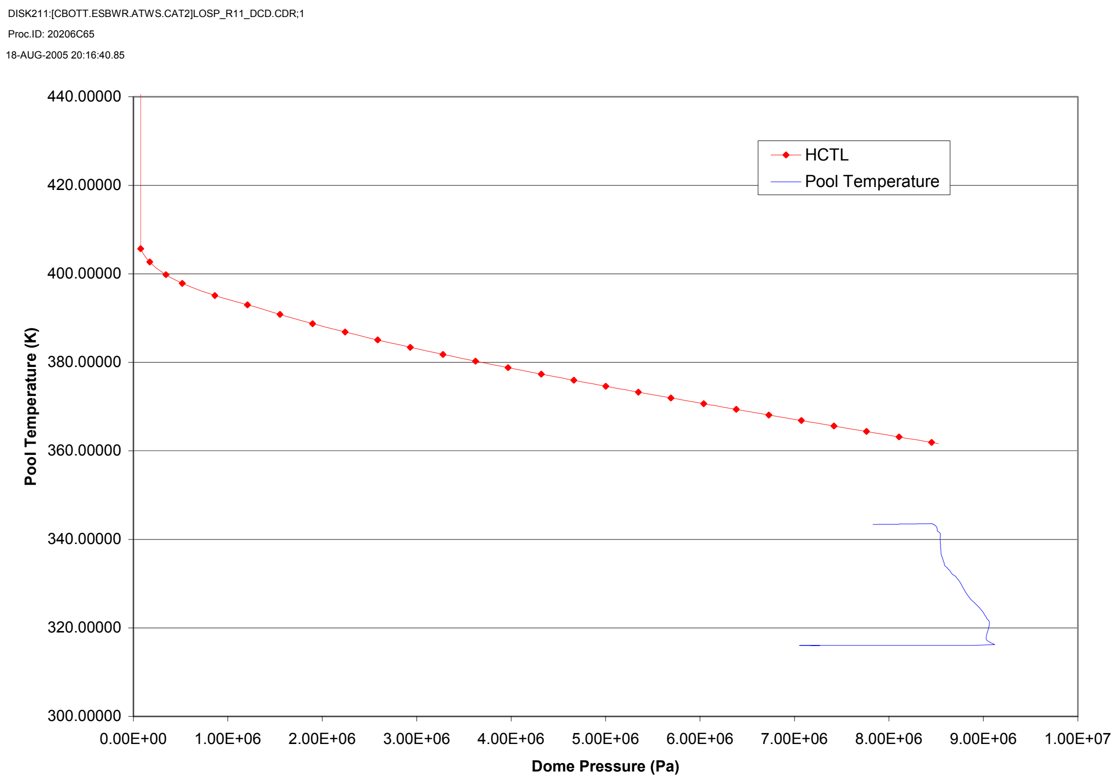


Figure 15.5-6d. Loss of Normal AC Power to Station Auxiliaries with Boron Injection

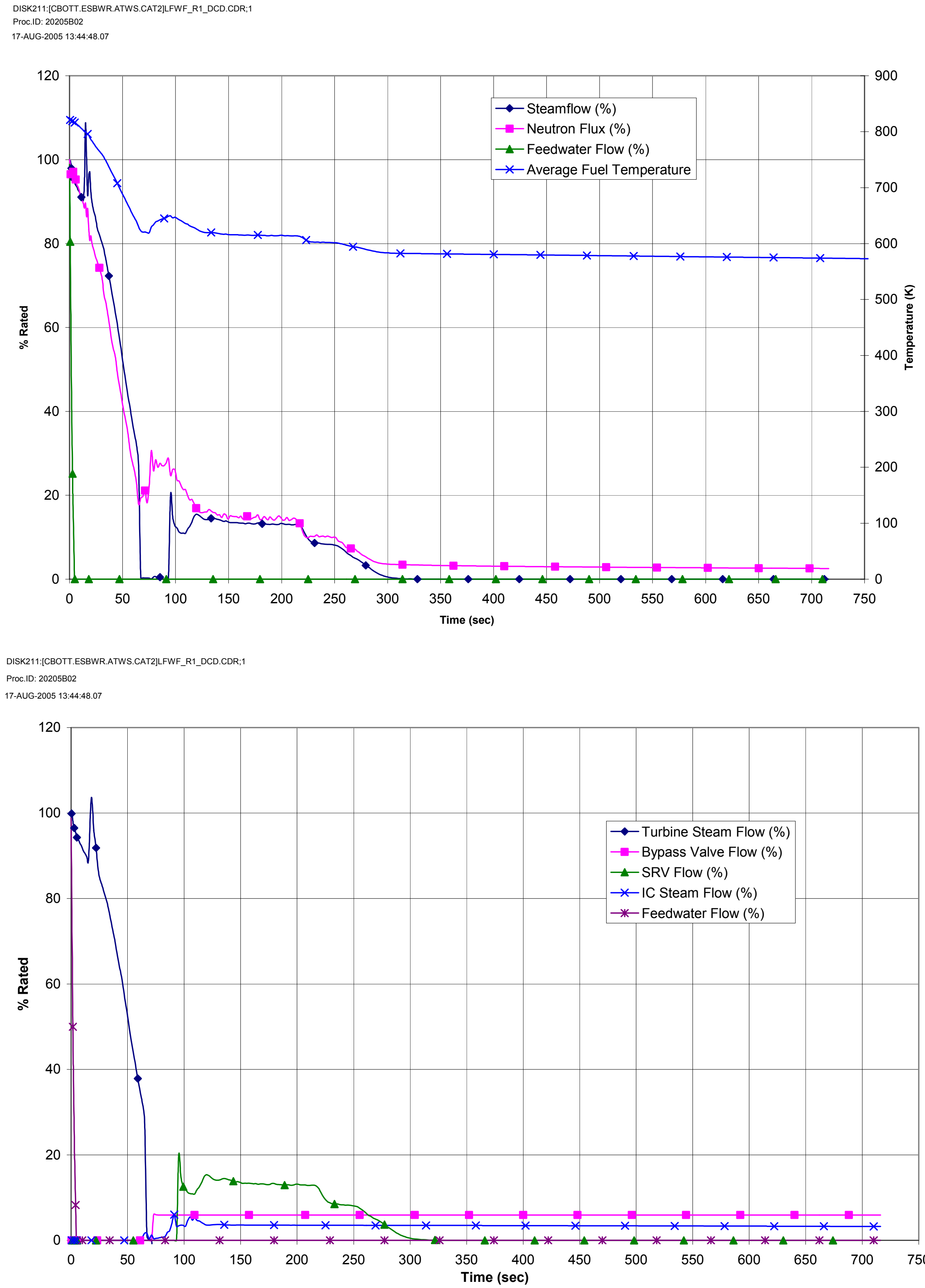
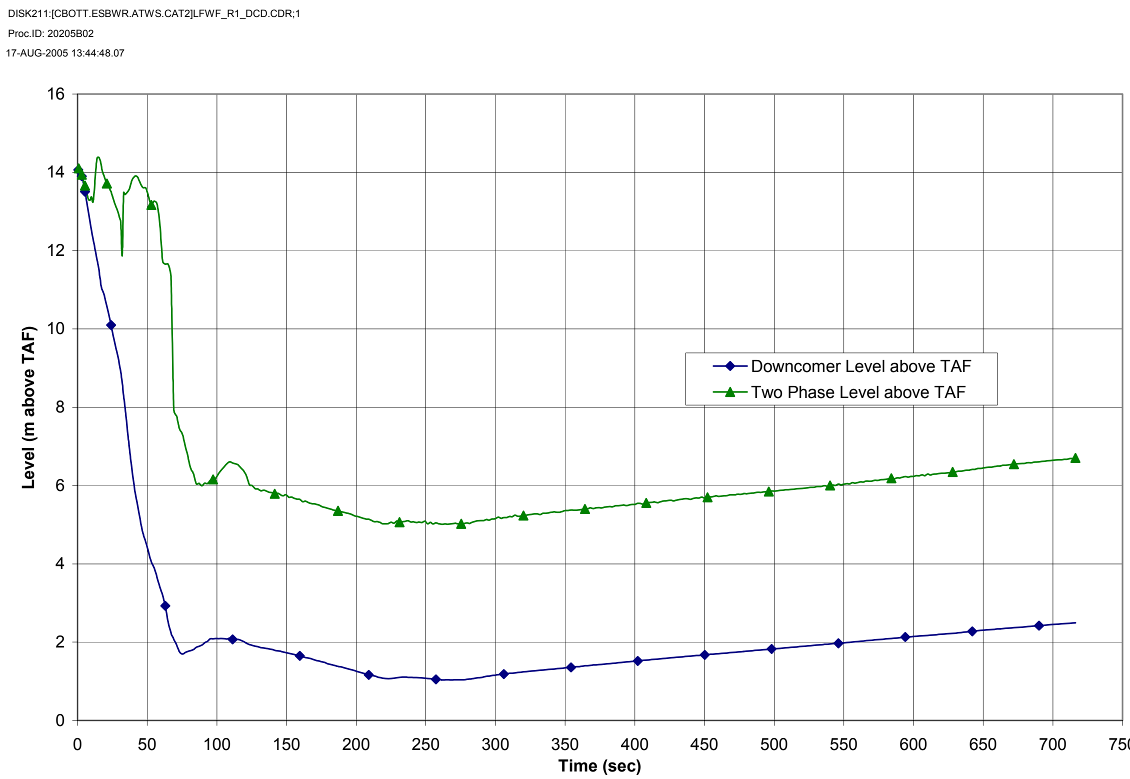


Figure 15.5-7a. Loss of Feedwater Flow with Boron Injection



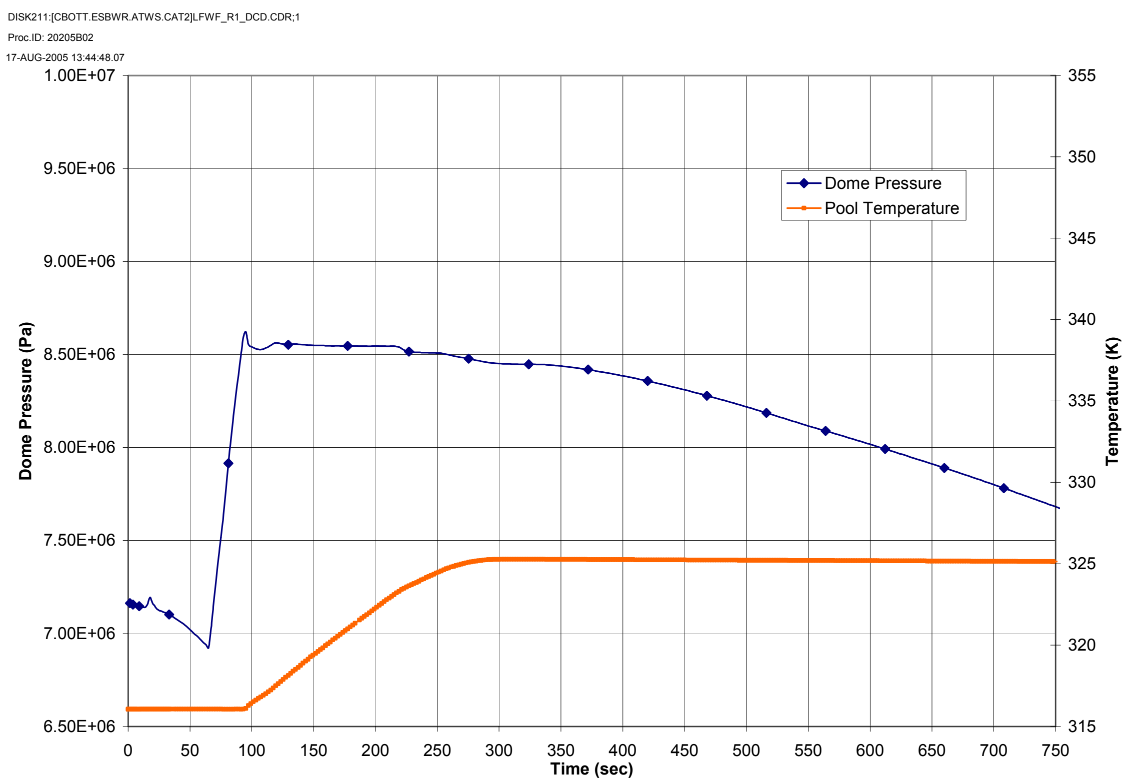
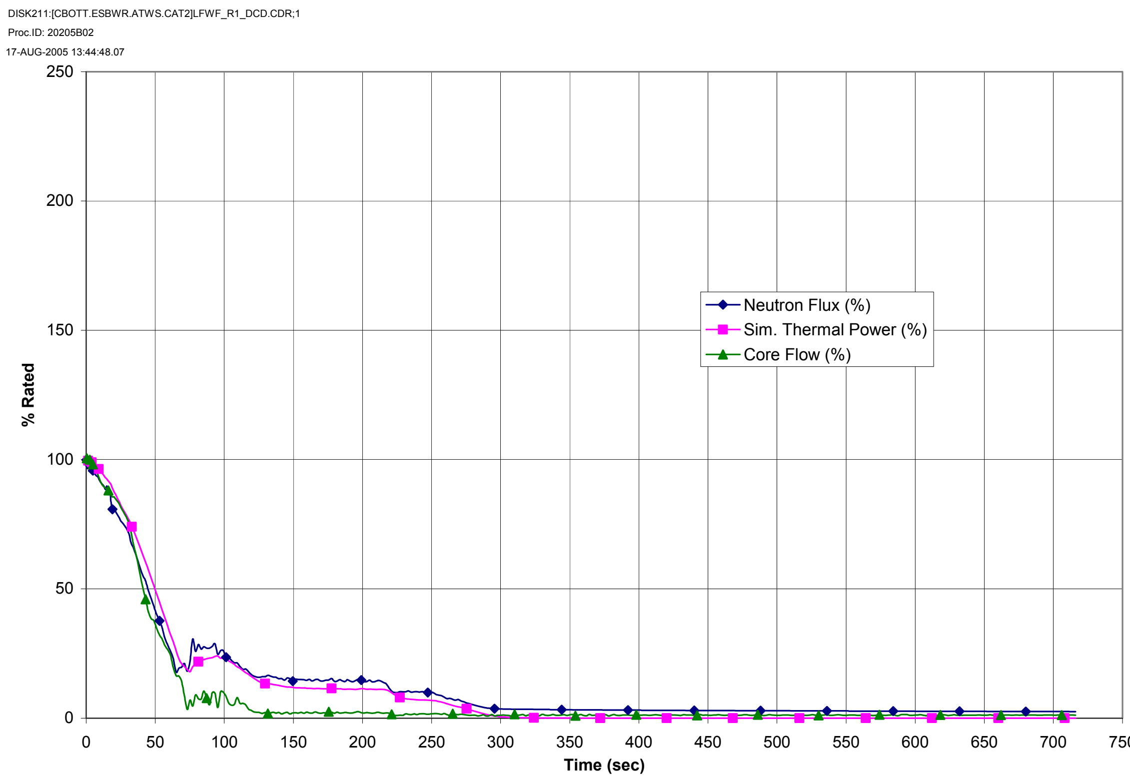


Figure 15.5-7b. Loss of Feedwater Flow with Boron Injection



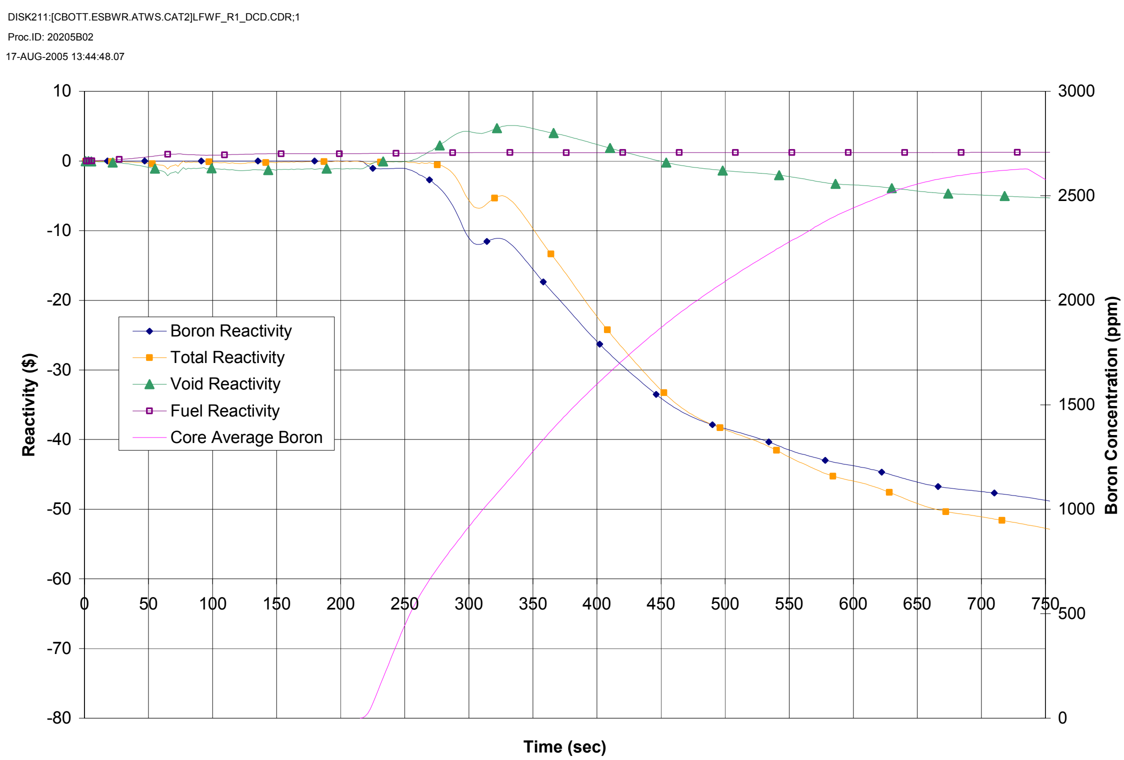


Figure 15.5-7c. Loss of Feedwater Flow with Boron Injection

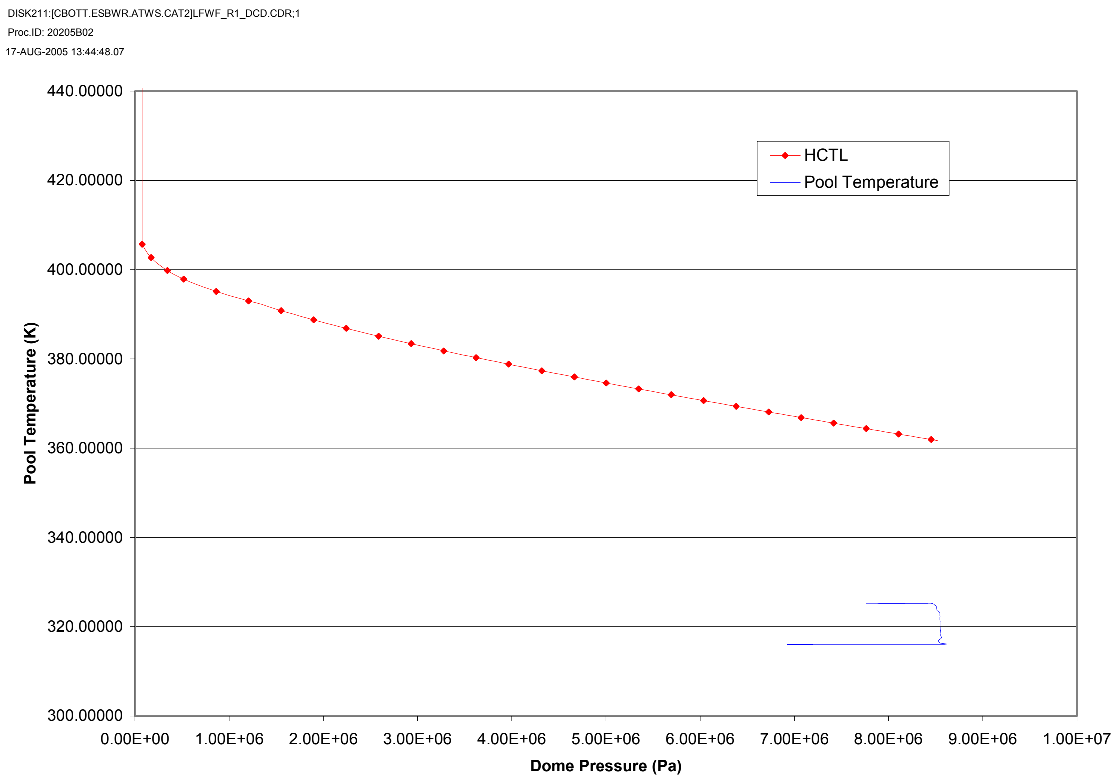
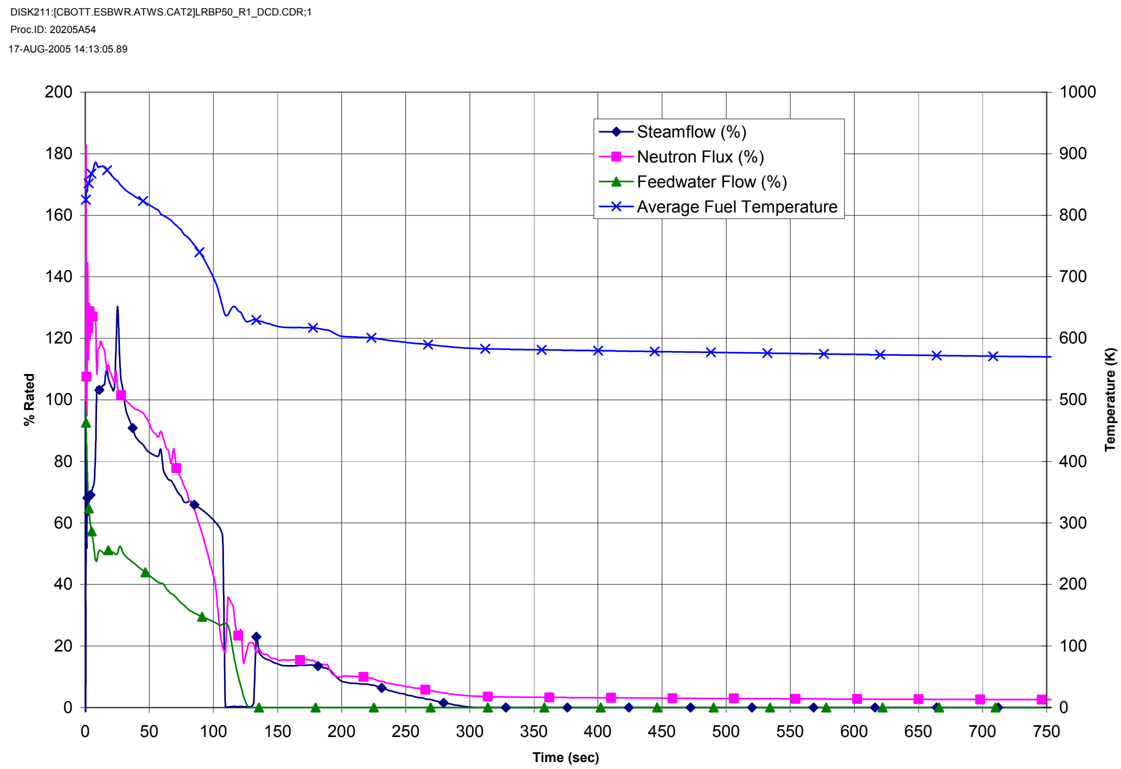


Figure 15.5-7d. Loss of Feedwater Flow with Boron Injection



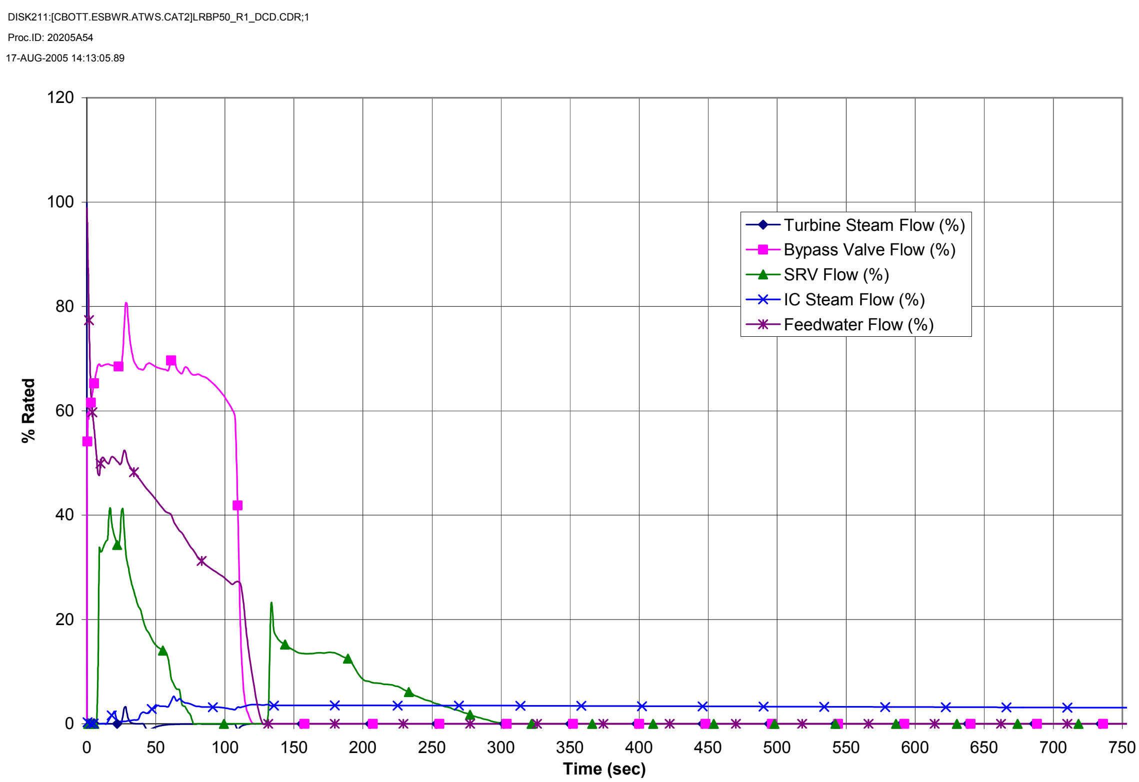
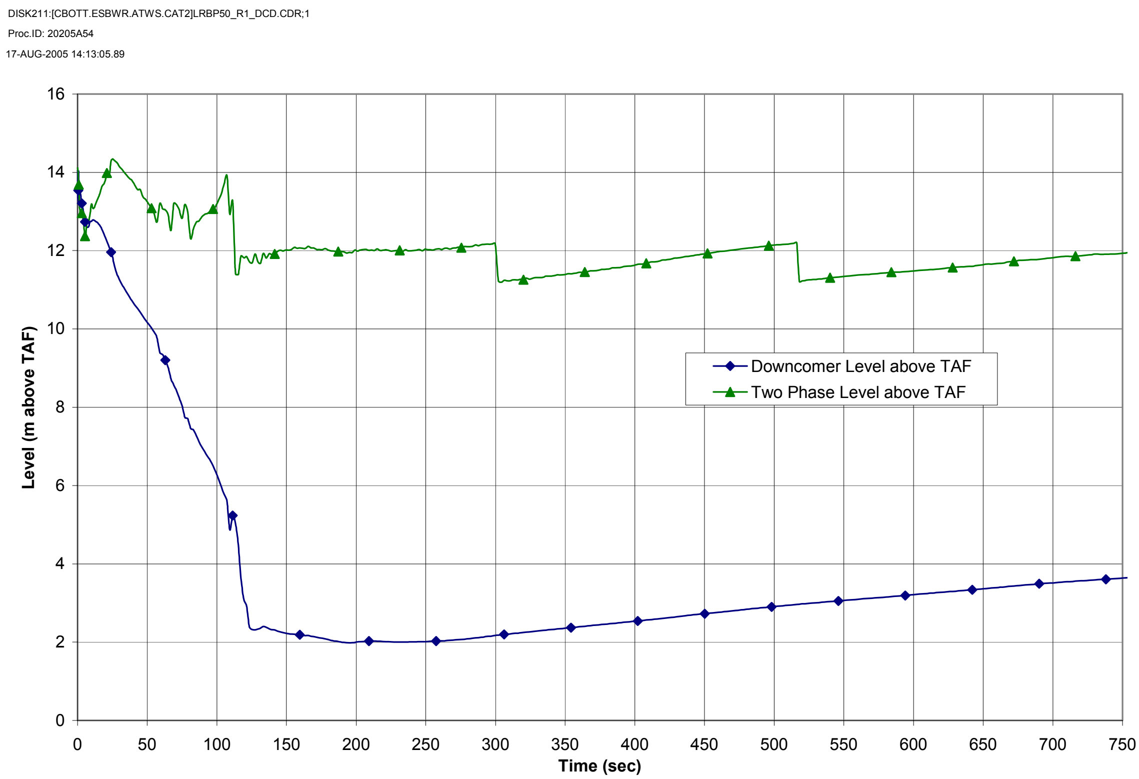


Figure 15.5-8a. Load Rejection with a Single Failure in the Turbine Bypass System with Boron Injection



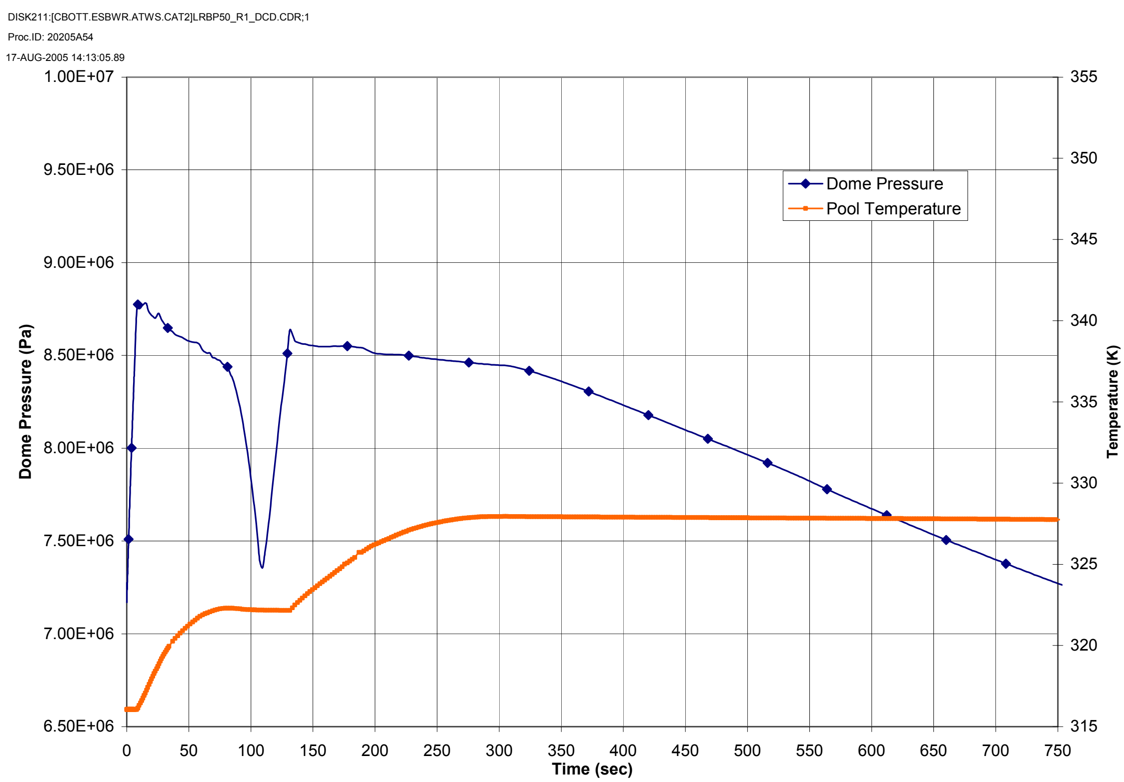
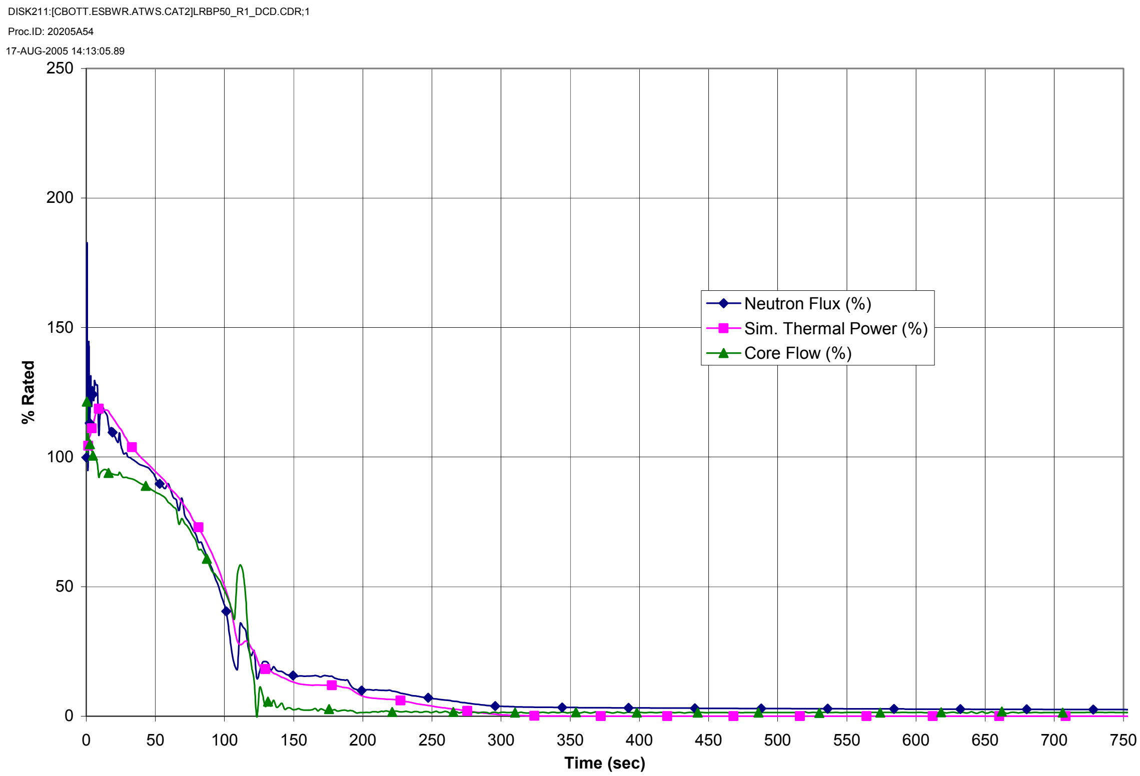


Figure 15.5-8b. Load Rejection with a Single Failure in the Turbine Bypass System with Boron Injection



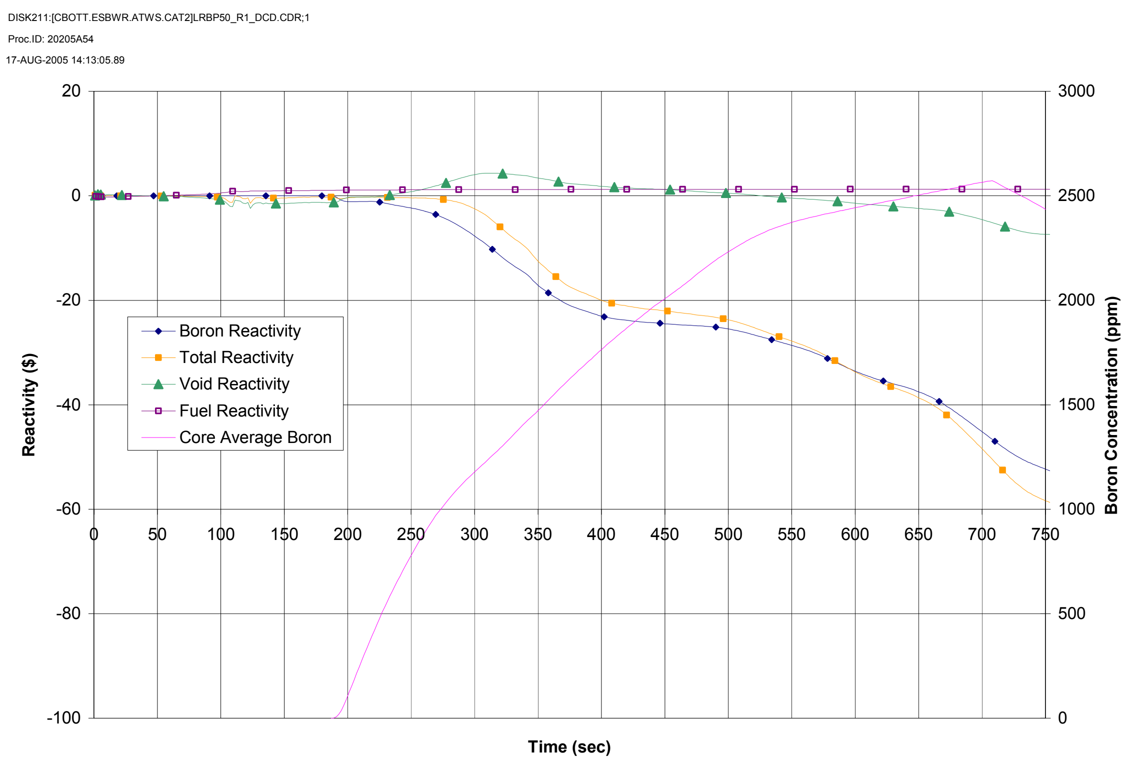


Figure 15.5-8c. Load Rejection with a Single Failure in the Turbine Bypass System with Boron Injection

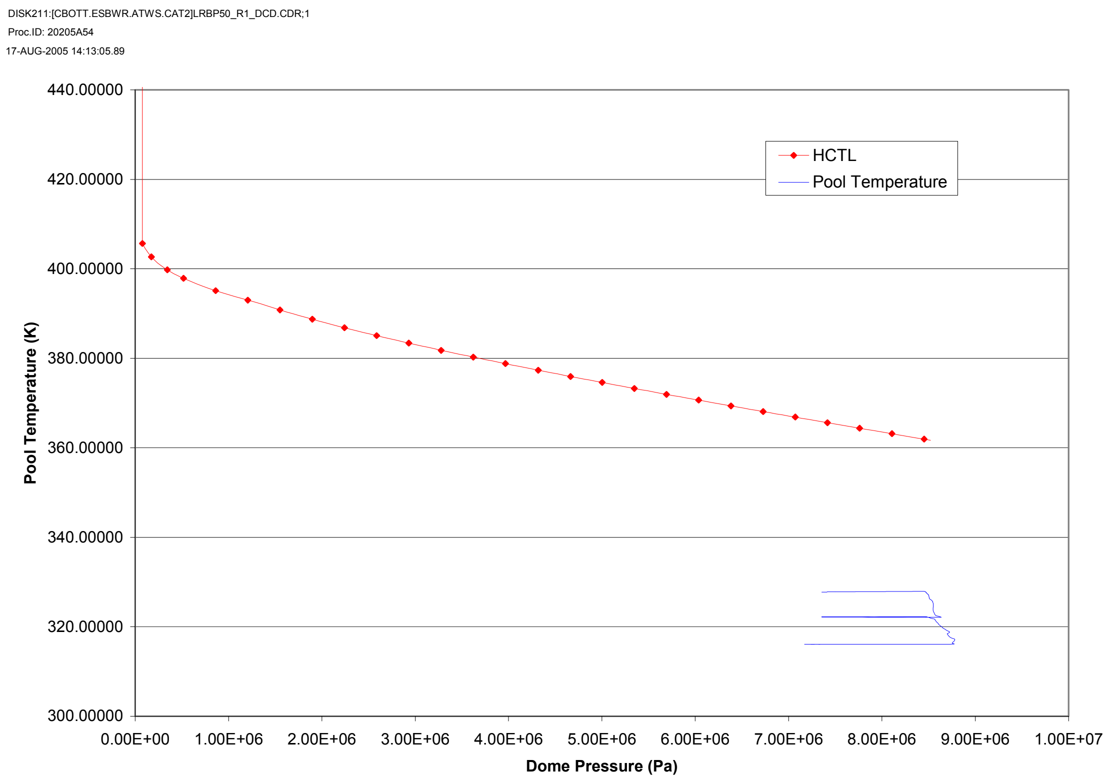


Figure 15.5-8d. Load Rejection with a Single Failure in the Turbine Bypass System with Boron Injection

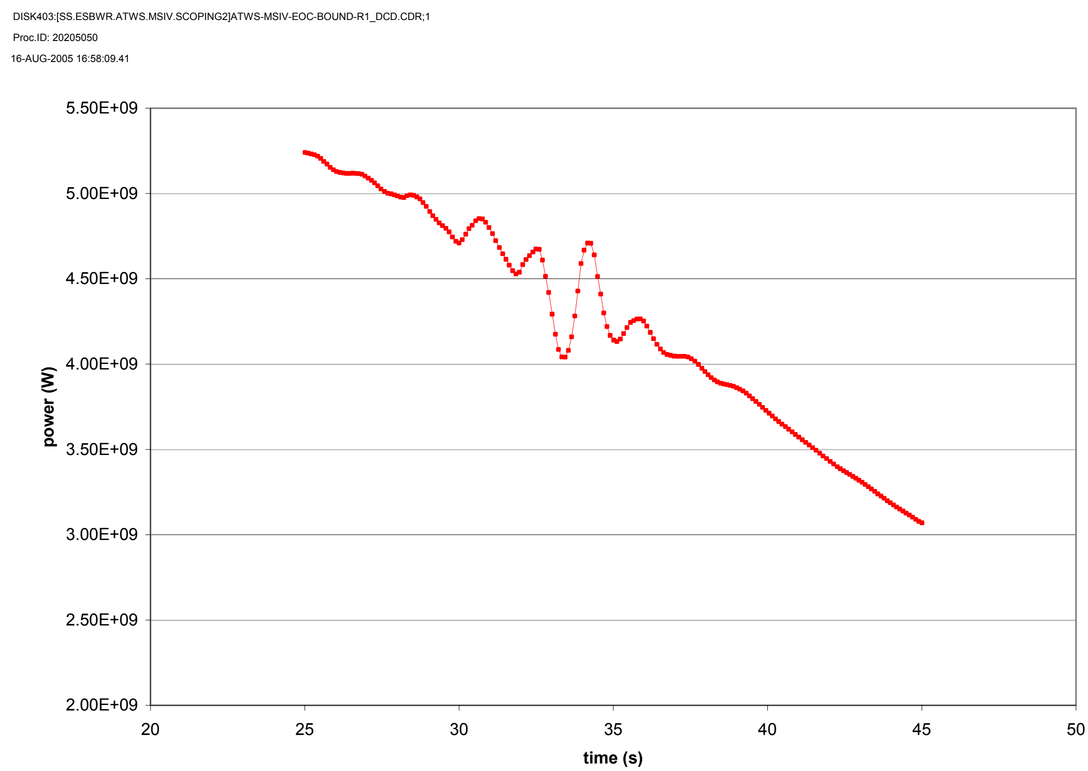


Figure 15.5-9. Core Stability during ATWS MSIV Closure Event

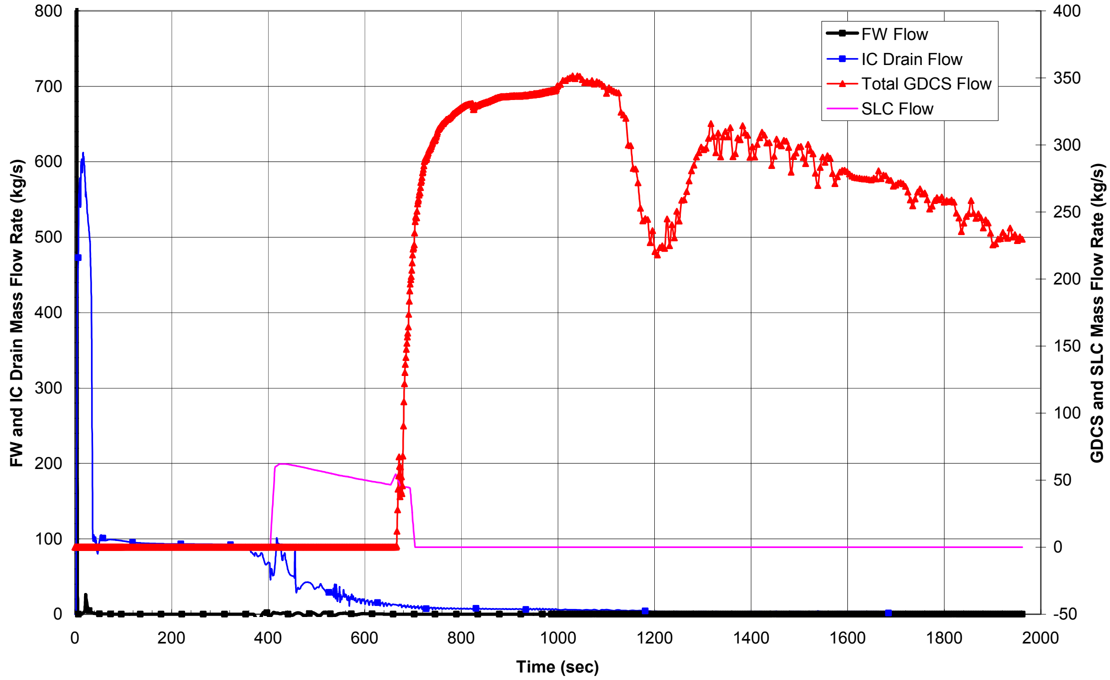


Figure 15.5-10. Vessel Inventory Makeup Flow Response for SBO

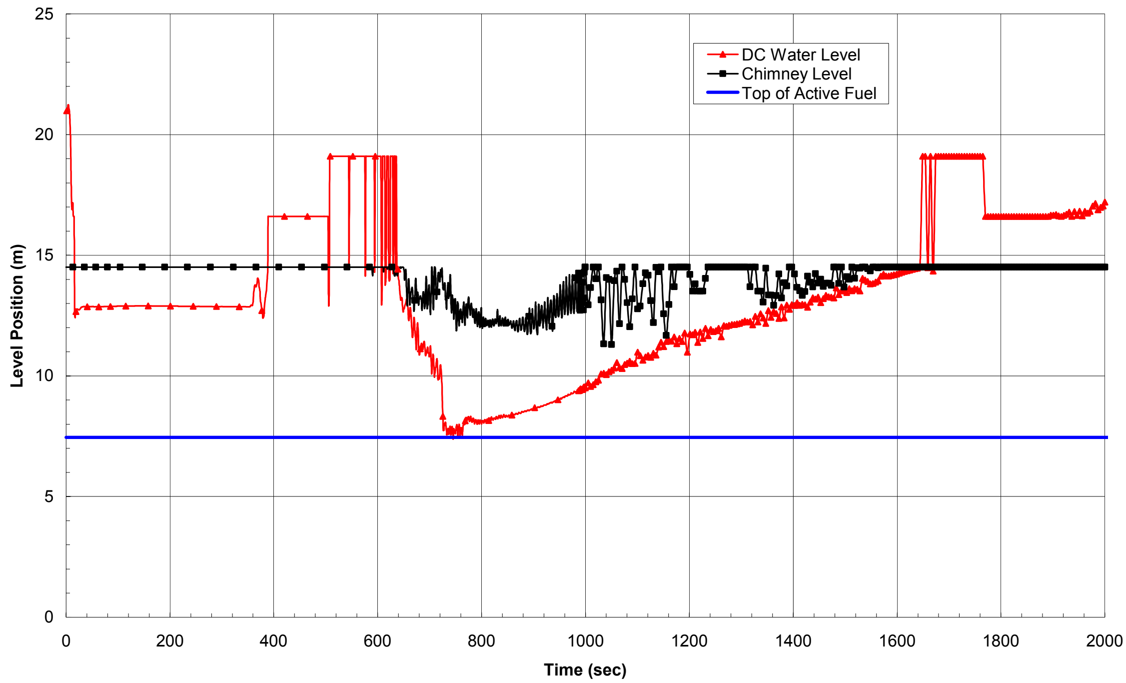


Figure 15.5-11. Water Level Response for SBO

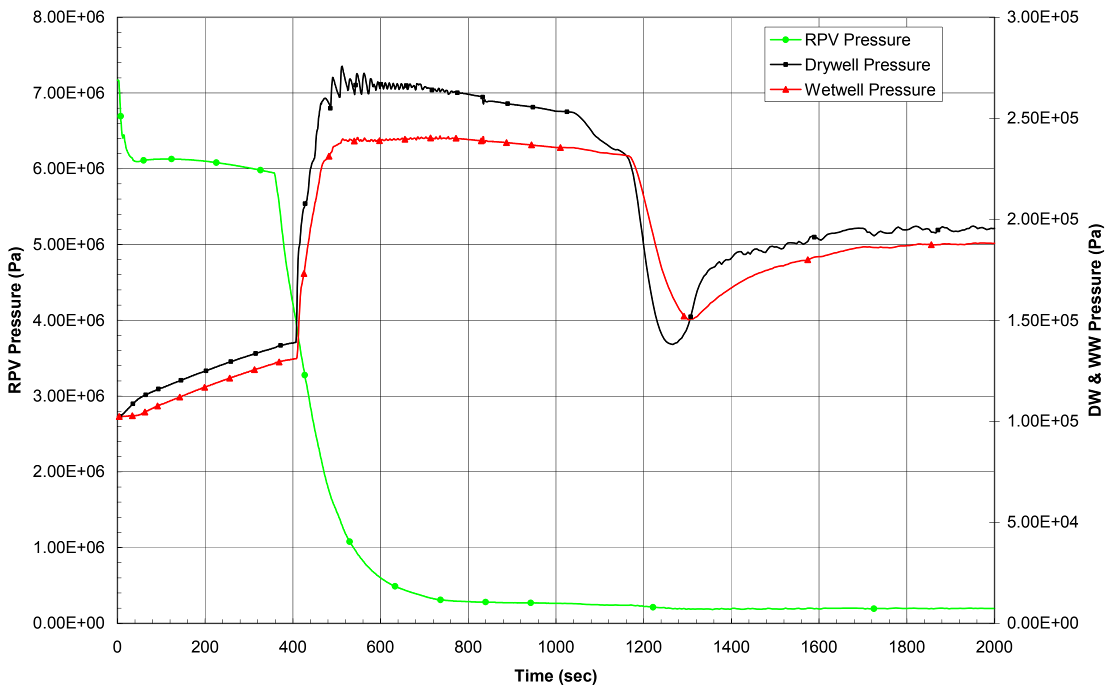


Figure 15.5-12. Vessel and Containment Pressure Response for SBO

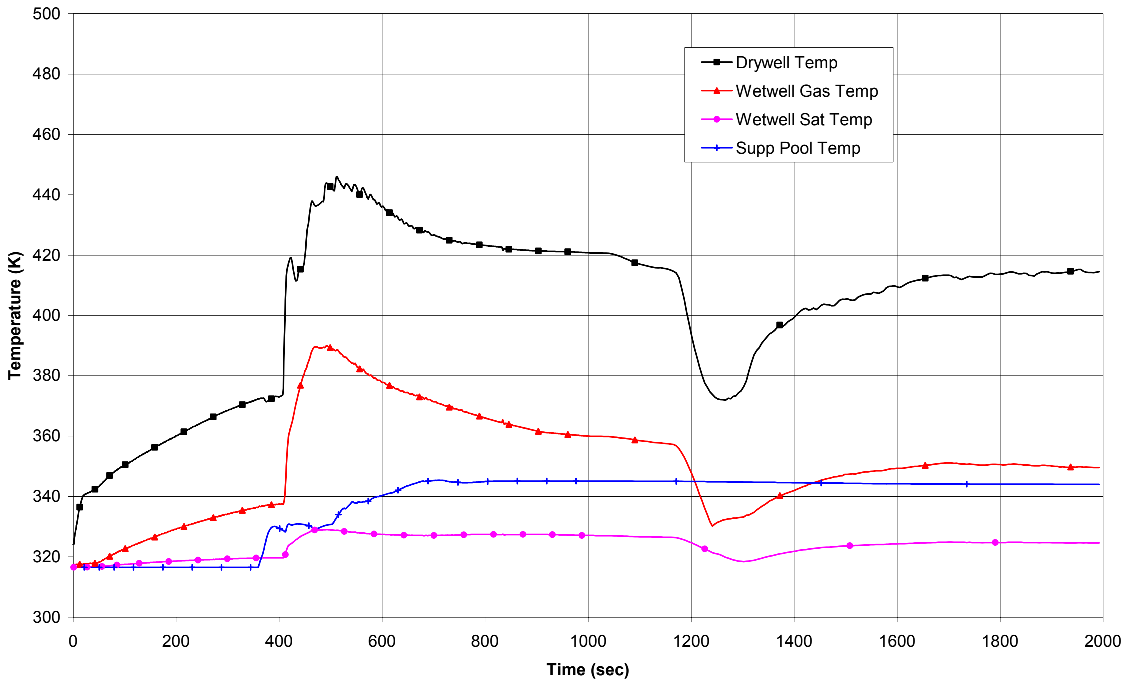


Figure 15.5-13. Containment and Suppression Pool Temperature Response for SBO

15A. EVENT PROBABILITY ANALYSES

This Appendix evaluates the annual event probabilities to determine if some specific events should be classified as Infrequent Events for the ESBWR. Classification of previous BWR designs may not be appropriate for the ESBWR because of the plant's advanced features and additional redundancies. The events that are being evaluated in this Appendix

- have historically been analyzed as Anticipated Operational Occurrences (AOOs), or
- are ESBWR design-unique.

The following discussion provides evaluations needed to determine the appropriate event classification of these events for the ESBWR.

Table 15A-2 summarizes the estimated probability of occurrence for each of the events analyzed in this Appendix.

15A.1 PRESSURE REGULATOR FAILURES

Reliability analyses have been performed on the ESBWR pressure regulator fault tolerant controller architecture. The analysis determines the expected frequency of the control system from failures of the pressure regulator. There are several possible adverse effects of such a failure:

- Simultaneous closure of all four turbine control valves (TCVs)
- Simultaneous opening of all TCVs and all turbine bypass valves (TBVs)

Closure of all four TCVs initiated by the simultaneous undetected failure of two pressure control system channels is defined as "Pressure Regulator Downscale Failure" (PRDF). In the event of two detected failures, a turbine trip is initiated to avoid a PRDF event, which is analyzed as a turbine trip with a failure of all turbine bypass valves (TTWFATBV).

There is no single failure that can lead to a PRDF and this event should be classified as an Infrequent Occurrence. The following discussion provides the basis for that classification.

15A.1.1 Pressure Regulator Downscale Failure

The ESBWR Steam Bypass and Pressure Control (SB&PC) System uses a triplicated digital control system. The SB&PC System controls TCVs and turbine bypass valves to maintain reactor pressure. No credible single failure in the control system results in a minimum demand to all Turbine Control Valves (TCVs) and bypass valves. A voter or actuator failure may result in an inadvertent closure of one turbine control valve or one turbine bypass valve if it is open at the time of failure. In this case, the SB&PC System senses the pressure change, commands the remaining control or bypass valves, if needed, to open, and thereby automatically mitigates the transient by maintaining reactor power and pressure.

No single failure causes the SB&PC System to issue erroneously a minimum demand to all TCVs and bypass valves. However, multiple failures might cause the SB&PC System to fail and issue a minimum demand. The SB&PC System design includes provision to mitigate the effects of this postulated multiple failure event. Diagnostics are conducted by comparison of internal digital TCV position demand signals to determine the failure of any output signals to TCVs. The

TCV demand output signal failure from the controller is considered as a channel failure. In the event of a detected failure of two channels of the triplicated control system, a turbine trip is automatically initiated. For this sequence of failures, the event is analyzed as a Turbine Trip Without Bypass (TTWFATBV). The simultaneous failure of two control processors, with at least undetected channel failure is called "pressure regulator downscale failure (PRDF)." It differs from the TTWFATBV because turbine trip is not automatically initiated.

15A.1.1.1 System Description

The elements of the Steam Bypass and Pressure Control System (SBPCS) are depicted in Figure 15A.1-1. The pressure sensor signals are subject to range limit checks to identify a fully failed sensor/signal. A selection logic is used to validate when all the three inputs are available and good. Upon failure detection of one signal, the validation logic automatically reverts to a high value gate (HVG) of the two remaining active signals.

The control system consists of three identical processing channels with necessary hardware and firmware. Means are provided to transfer data between processing channels. To avoid processing channel output divergence, the processing channels compare and vote on calculated integrator state variables. The signal voting and interprocessor communication is implemented to assure that no more than one sample period delay occurs between sampling the inputs and using them in the processor calculations.

15A.1.1.2 Analysis

The assumption for this analysis is that TCV demand output signal failures from two channels has occurred, and that one or more of these failures is not detected by the diagnostics. The result is closure of all four TCVs.

15A.1.1.2.1 Analytical Conditions

Mean-time-to-repair is shown on Table 15A-3, item 1.

It was assumed that the failure rates represent failures that result in upscale failure of the output.

Detected failure of two channels causes a turbine trip. The plant condition is one in which the turbine is isolated, but all turbine bypass valves are open. Steam flow is terminated when the Main Steam Isolation Valves close.

The first failure of a channel is evaluated as shown on Figure 15A-1-2. The result is for the failure of one (and only one) of the three channels in a year. The probability of this failure not being detected is based on the failure of a complex circuit (Table 15A-3, item 2).

The Fault Tree for a second failure when the first failure is detected is shown on Figure 15A.1-3. The second channel failure occurs during the period when the first channel is being repaired.

Figure 15A.1-4 shows the Fault Tree for a second failure when the first failure is undetected. The first failure is assumed to occur, on average, at six months and the second failure can occur anytime during the subsequent six month period.

15A.1.1.2.2 Approach

Using the system block diagram of Figure 15A.1-1, the event trees (Figures 15A.1-2 through 15A.1-5) were constructed to show the failure paths which could result in pressure regulator downscale failure. Basically, there are two ways that all four turbine control valves can be closed "simultaneously": (1) failure of combinations of all four valves, or (2) failure of any two of the three channels, with the two possible combinations of at least one of the two failures not detected. When a turbine trip is initiated because two sensor failures have been detected, then a PRDF event has been avoided. This sequence is analyzed as a turbine trip with failure of all bypass valves.

The logic equations (Table 15A-1) were written from the event trees and after simplification were evaluated for the frequencies identified in the introduction.

15A.1.1.3 Results

The frequency of inadvertent closure of all 4 TCVs, initiated by failure of the pressure regulator, is found to be extremely low so that the event should be treated as an Infrequent Event. Inadvertent closure of all four TCVs is shown on Figure 15A.1-5 and Table 15A-2.

15A.1.2 Pressure Regulator Upscale Failure

No credible single failure in the Steam Bypass and Pressure Control System (SBPCS) results in a maximum demand to all actuators for all TCVs and turbine bypass valves (PRUF). A voter or actuator failure may result in an inadvertent opening of one TCV or one turbine bypass valve. However, multiple failures might cause the SB&PC System to fail and issue a maximum demand.

Opening of all four TCVs and all turbine bypass valves initiated by the simultaneous failure of two pressure control channels, at least one of which is undetected, is defined as "Pressure Regulator Upscale Failure" (PRUF). Note that if both failures are detected, then the turbine is tripped and the PRUF event has been terminated.

There is no single failure that can lead to a PRUF and this event should be classified as an Infrequent Occurrence

15A.1.2.1 System Description

Refer to Subsection 15A.1.1.1 for a discussion of the Steam and Bypass Control System.

Diagnostics are conducted by comparison of internal digital TCV position demand signals to determine the failure of any output signals to TCVs. The TCV demand output signal failure from the controller is considered as a channel failure. In the event of two detected failures, a turbine trip is initiated which mitigates the PRUF event. The plant condition is one in which the turbine is isolated, but all turbine bypass valves are open. Steam flow is terminated when the Main Steam Isolation Valves close. The simultaneous failure of two control processors, with at least one of the failures undetected, is called "pressure regulator upscale failure (PRUF)."

SB&PC functional logic is performed by a triple-redundant, microprocessor based FTDC. The FTDC consists of three identical processing channels working in parallel to provide fault-tolerant operation.

The FTDC is capable of meeting the following requirements:

- Dual channel failure rate ≤ 10^{-6} / hour
- Mean time to repair – Table15A-3, item 1.
- Success rate of first channel failure detection ≥ 99.99 %
- Success rate of 2nd channel failure detection ≥ 90 % (in conjunction with TCS-EHC)

15A.1.2.2 Analysis

The assumption for this analysis is that TCV demand output signal failures from two channels has occurred, and that one or more of these failures is not detected by the diagnostics. The result is opening of all four TCVs and all Turbine Bypass Valves.

15A.1.2.2.1 Analytical Conditions

It was assumed that the failure rates represent failures that result in high output.

Mean-time-to-repair is shown on Table 15A-3, item 1.

Failure rates for electronic modules are estimated, based on anticipated complexity of the circuit functions as shown on Table 15A-3, item 2.

The assumption for this analysis is that TCV demand high output signal failures from two channels have occurred, one or both of these failures is not detected by the diagnostics. As a result, the TBCS generates an open signal for all four TCVs and all of the Turbine Bypass Valves.

15A.1.2.2.2 Approach

Using the system block diagram of Figure 15A.1-1, event and fault trees (Figures 15A.1-2 through 15A.1-4 and 15A.1-6) were constructed to show the failure paths which could result in open failure of all turbine control and bypass valves. The only credible way that all of these valves can be opened "simultaneously" is by failure of any two of the three channels, with at least one of the two channel failures not detected. If a turbine trip is initiated because two sensor failures have been detected, then the PRUF event has been avoided. Turbine bypass valves remain open and are isolated when the MSIVs close on a low RPV water level signal.

The logic equations (Table 15A-1) were written from the event trees and after simplification were evaluated for the failure frequencies.

The frequency of inadvertent closure of all four TCVs initiated by failure of the pressure regulator is found to be extremely low so that the event can be treated as an Infrequent Event.

15A.1.2.3 Results

The frequency of inadvertent openings of all four TCVs and all TBVs, initiated by failure of the pressure regulator, is found to be extremely low so that the event should be treated as an Infrequent Event. The Event Tree for Inadvertent opening of all four TCVs and TBVs is shown on Figure 15A.1-6 and the results of the analysis are shown on Table 15A-2.

15A.2 INITIATING EVENTS WITH 100% TURBINE BYPASS FAILURE

There is no single failure that can lead to either a generator load rejection with Failure of all turbine bypass valves (GLRWFATBV) or a turbine trip with failure of all turbine bypass valves (TTWFATBV). Both of these events should be classified as an Infrequent Occurrence. The following discussion provides the basis for this classification.

15A.2.1 Turbine Trip with Failure of All TBVs

A variety of turbine or nuclear system malfunctions initiate turbine trips. After the main turbine is tripped, turbine bypass valves are opened in their fast opening mode by the Steam Bypass and Pressure Control System (SB&PC). Any single failure can only cause one bypass valve to fail to open on demand. Multiple failures are required to cause all bypass valves to fail to open on demand. The frequency of occurrence for this event is significantly less than 0.01 events per year. As a result, this event should be classified as an Infrequent Occurrence. The following evaluation provides the basis for this conclusion.

15A.2.1.1 System Description

The elements of the SB&PC control system are depicted in Figure 15A.2-5. The pressure sensor signals are subject to range limit checks to identify a fully failed sensor/signal. A selection logic is used to validate when all the three inputs are available and good. Upon failure detection of one signal, the validation logic automatically reverts to a high value gate (HVG) of the two remaining active signals.

The control system consists of three identical processing channels with necessary hardware and firmware. To avoid processing channel output divergence, the processing channels compare and vote on calculated integrator state variables. Signal voting and interprocessor communication is implemented to assure that no more than one sample period delay occurs between sampling the inputs and using them in the processor calculations.

15A.2.1.2 Analysis

This analysis considers two means by which the plant can experience a turbine trip with failure of all turbine bypass valves.

Turbine Control Valve (TCV) demand output signal failure from two channels can occur resulting in closure of all four TCVs.

Turbine trip occurs independently but in close time proximity to failure of all turbine bypass valves. TBV failure is caused by hydraulic pump failure, or loss of the hydraulic power unit power.

Loss of condenser vacuum will result in a turbine trip and is analyzed as a separate event. It is not included in the Event Trees for TTWFATBV.

15A.2.1.2.1 Analytical Conditions

It was assumed that the failure rates represent failures that result in loss of output.

Mean-time-to-repair is shown on Table 15A-3, item 1.

Failure rates for electronic modules are estimated, based on anticipated complexity of the circuit functions as shown on Table 15A-3, item 2.

15A.2.1.2.2 Approach

The first way a TTWFATBV event is initiated because two sensor failures have been detected. The turbine is tripped automatically and the TBVs do not receive an open signal because of the downscale failure of the regulators. If a sensor failure is not detected, then the event outcome is the same. TCV closure with a closed signal demand on the TBVs will still occur because of the downscale failure of the sensors. The system block diagram of Figure 15A.1-1 and the event trees (Figures 15A.1-2 through 15A.1-5) were constructed to show the failure paths which could result in channel failures in the SB&CS that would result in a turbine trip and close demand signal to all of the TBVs.

There are only two differences between a detected and non-detected failure. First, if the first sensor failure is detected, then there is a probability of repair before the second sensor fails. If the first failure is undetected, then that failure is assumed to occur, on average at the midpoint of the year. The second failure could then occur anytime in the following six months. The Event Tree for this condition is shown on Figure 15A.1-4. Second, the closure time of the TCVs will be more gradual because there is no automatic trip of the turbine by closure of the turbine stop valves. Figure 15A.2-1 shows the Event Tree for this sequence of failures.

The second way a TTWFATBV event can occur is by failure of the hydraulic power unit supplying control to the turbine and turbine bypass valves. The dominant failures mechanisms were judged to be from loss of redundant hydraulic pumps or from loss of redundant inverter power supplies. The piping and manual isolation valves between the TCV HCUs were not considered as a likely failure mechanism. The failure analysis is based on the failure rates for various system components as shown on Table 15A-3. TBV is controlled by a three-coil servo valve. Each FTDC provides the TCV position demand signal to the associated analog servo control circuit to provide analog current signal to each coil of the TCV. Actual voting of the three coils takes place at the three coil analog circuitry. Loss of one analog current signal to the coil shall not cause the valve to close.

The simultaneous failure of all 12 valves in proximity to an independent event that caused the turbine to trip was not included in the failure analysis. This combination of events was judged to be of a low enough probability that it did not warrant a numerical evaluation. The Event Trees were constructed using the failure rates for Table 15A-3 items 10, 11 and 16. Figures 15A.2-2 and 15A.2-3 show the Event Tree for this sequence of failures.

The logic equations (Table 15A-1) were written from the event trees and after simplification were evaluated for the frequencies identified in the introduction.

15A.2.1.3 Results

The frequency of inadvertent closure of all Turbine Trip with Failure of All Bypass Valves initiated by failure of the pressure regulator, is found to be low. The event should be treated as an Infrequent Event. Fault Tree for TTWFATBV is shown on Figure 15A.2-4 and the results of the analysis are shown on Table 15A-2.

15A.2.2 Generator Load Rejection with Failure of All TBVs

There is no single failure that can lead to a Generator Load Rejection with Failure of All Turbine Bypass Valves (GLRWFATBV).

In Subsection 15A.2.1 the failure rate for a Turbine Trip With Failure Of All Bypass Valves was shown to support classification of TTWFABV as an Infrequent Event. The failure rate for the Generator Load Rejection is lower than for the TTWFABV. Therefore, GLRWFATBV should be classified as an Infrequent Event.

15A.2.2.1 System Description

Fast closure of the turbine control valves (TCVs) is initiated whenever electrical grid disturbances occur which result in significant loss of electrical load on the generator.

After sensing a significant loss of electrical load on the generator, the TCVs are commanded to close rapidly. At the same time, the turbine bypass valves are signaled to open in the "fast" opening

The elements of the SB&PC control system are depicted in Figure 15A.1-1. The pressure sensor signals are subject to range limit checks to identify a fully failed sensor/signal. A selection logic is used to validate when all the three inputs are available and good. Upon failure detection of one signal, the validation logic automatically reverts to a high value gate (HVG) of the two remaining active signals.

The control system consists of three identical processing channels with necessary hardware and firmware. To avoid processing channel output divergence, the processing channels compare and vote on calculated integrator state variables. Signal voting and interprocessor communication is implemented to assure that no more than one sample period delay occurs between sampling the inputs and using them in the processor calculations.

Each TBV receives demand signals from, and provides valve position feedback signals to the Steam Bypass and Pressure Control System (SBPC). Hydraulic power for TBV actuation is provided from the main turbine hydraulic power unit.

The Turbine Control System consists of one hydraulic power unit and microprocessor-based digital electronic turbine controls. The unit includes two complete pumping systems.

The TBS (Main Steam System), consists of a chest that is connected to the main steamlines upstream of the turbine stop valves, and dump lines that separately connect each bypass valve outlet to one condenser shell.

The TBS system is powered from separate nonsafety-related divisional busses. The worst case single failure would result in a loss of no more than 50% of bypass capacity.

One valve chest is provided and houses three individual bypass valves. Each bypass valve is an angle body type valve operated by hydraulic fluid pressure with spring action to close. The valve chest assembly includes hydraulic supply and drain piping, three hydraulic accumulators (one for each bypass valve), servo valves, fast acting servo valves, and valve position transmitters.

The turbine bypass valves are operated by the turbine hydraulic fluid power unit. High-pressure hydraulic fluid is provided at the bottom valve actuator and drained back to the fluid reservoir.

The turbine bypass valves are opened by redundant signals received from the Steam Bypass and Pressure Control System whenever the actual steam pressure exceeds the preset steam pressure by a small margin. This bypass demand signal causes fluid pressure to be applied to the operating cylinder, which opens the individual valves. As the bypass demand increases, additional bypass valves are opened, dumping the steam to the condenser. The bypass valves are equipped with fast acting servo valves to allow rapid opening of bypass valves upon turbine trip or generator load rejection.

The bypass valves automatically trip closed whenever the vacuum in the main condenser falls below a preset value. The bypass valves are also closed on loss of electrical power or hydraulic system pressure. The bypass valve hydraulic accumulators have the capability to stroke the valves at least three times should the hydraulic power unit fail.

Low main condenser vacuum inhibits steam bypass valve opening and also initiates a turbine trip. The main steam isolation valves and the turbine bypass valves are closed (or inhibited from opening) as pressure increases toward a complete loss of vacuum. This action prevents overpressurization of the condenser shell. However, after turbine/generator trips from loss of offsite or station auxiliary power or from loss of condenser cooling, a minimum specified time of bypass operability is maintained by the control system.

15A.2.2.2 Analysis

A Generator Load Rejection with Failure of All Turbine Bypass Valves (GLRWFATBV) is a very unlikely event for the following reasons.

- The triplicated redundant Steam Bypass and Control System is very reliable, as shown in Section 15A.1.
- The motive power for the TBVs comes from the turbine hydraulic power unit, which is also highly reliable. Loss of the hydraulic power unit will result in a turbine trip. This event is a TTWFATBV not a Generator Load Rejection event.
- The TBS is designed to provide short-term operation on loss of power, loss of condenser vacuum, turbine trip and generator trip before closing the TBVs. Turbine trip will occur before the loss of turbine bypass capability.

15A.2.2.2.1 Analytical Conditions

Mean-time-to-repair is shown on Table 15A-3, item 1. Failure rates for electronic modules are estimated, based on anticipated complexity of the circuit functions as shown on Table 15A-3, item 2.

The frequency of a generator load rejection is identified on Table 15A-3, item 12.

15A.2.2.2.2 Approach

Using the system block diagram of Figure 15A.1-1, the event trees (Figures 15A.1-2 through 15A.1-5) were constructed to show the failure paths which could result in SB&CS channel failures that would result in loss of function of all of the turbine bypass valves.

The logic equations (Table 15A-1) were written from the event trees and after simplification were evaluated for the frequencies identified in the introduction.

When a turbine trip is initiated because two sensor failures have been detected, then the event has becomes a TTWFATBV.

The probability of occurrence of a GLRWFATBV is smaller than a TTWFATBV for the following reasons:

- The Initiating Event probability is lower than for a TT (see Table 15A-3, items 12 and 13).
- Failures in the SB&CS that results in a loss of all TBVs also initiates a turbine trip and terminates the Load Rejection Event unless at least one of the channel failures is undetected. This contingency is a very low probability occurrence as shown on Figure 15A.2-1
- Failure in the EHC motive power or control to the TBVs will also cause a turbine trip.

The independent, failure of TBV demand output signal from two channels of SB&PC is a very small number and supports the conclusion that the event should be classified as an Infrequent Event. Note that the failures have to occur in proximity of the load rejection event and the failure probability is small. Refer to Figure 15A.1-3.

15A.2.2.3 Results

The frequency of a Generator Load Rejection event in proximity to a failure of all TBVs, is less than the occurrence of a TTWFATBVs and is extremely low. The event should be treated as an Infrequent Event. This conclusion is reflected on Table 15A-2.

15A.3 FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND

This event is analyzed in the safety analysis as the runout of all four feedwater pumps. High water level turbine trip and feedwater flow runback are initiated early in the event. Reactor scram limits the neutron flux and turbine bypass system limits primary system pressure.

There is no single failure that can lead to this condition and this event should be classified as an Infrequent Occurrence. The following discussion provides the basis for that classification.

15A.3.1 System Description

15A.3.1.1 Feedwater System

The Feedwater System consists of four motor-driven feedwater pumps powered by adjustable speed drives. Each pair of main feedwater and booster pumps is driven by the same motor. The Feedwater Control System (FWCS) supplies the demand signal to the feedwater pump ASD.

Maximum feedwater runout capacity is 155%.

15A.3.1.2 Feedwater Control system

FWCS provides logic for controlling the supply of feedwater flow to the reactor vessel in response to automatic or operator manual control signals. This control maintains reactor water level within predetermined limits for all operating conditions including startup.

The FWCS is a Fault-Tolerant Digital Controller (FTDC), which is a triplicated, microprocessor-based controller that executes the control software and logic required for reactor level control and other FWCS functions. The system contains three identical processing channels (operating in parallel) that receive inputs from other systems and issue actuator and speed demands, process measurement data, interlock and trip signals (refer to Figure 15A.3-1). The FTDC issues actuator demand signals to the Low Flow Control Valve (LFCV) and a speed demand signal to the Feedwater Pump variable speed controllers.

The plant will operate normally with the failure, shutdown, isolation, repair or testing of any single channel of the triplicated controller. To insure plant availability, there are two channels of the triplicated controller available at all times. Input signals to FWCS are redundant and conveyed to FWCS processors via a highly reliable multiplexing system. This minimizes problems associated with faulty input signals. The electrical input power for the FTDC is from separate uninterruptible power sources to minimize adverse effects due to loss of a single electrical power source.

The basic water level control modes for FWCS are single-element and three-element level control. The master level controller generates a demand signal available to the actuators and controllers.

The final FWCS actuator speed demand signals control demand sent to the actuator is validated. The proper operation of the voter is validated by returning the final voter output to the FTDCs in order to detect a failure.

On detecting a voter output error, a lockup command is sent to a discrete lockup voter. On a two-out-of-three vote, the lockup voter issues a command to lockup the actuator with the defective voter output and send an alarm to alert the operator. The demand on the remaining feedwater pumps will decrease and offset the increased flow from one pump.

15A.3.2 Analysis

15A.3.2.1 Analytical Conditions

Failure rate for an actuator is assumed to be .0088/year.

Mean-time-to-repair (MTTR) is shown on Table 15A-3, item 1. Failure rates for electronic modules are estimated, based on anticipated complexity of the circuit functions is shown on Table 15A-3, item 2.

Failure of all four actuators would be required to cause the design basis FCFWMD. Because there are four pumps failure of only three pumps at 155% of rated flow would provide 155%(75%) = 116% of rated flow. Only a mild excess feedwater flow condition would occur as long as the fourth pump responded as designed and ran back flow.

The assumption for this analysis is that the first failure is an actuator because this is a more likely failure than an electronic module (Figure 15A.3-2). It is conservatively assumed that the failure of a single electronic module can cause generation of a high demand signal to the remaining three pumps.

15A.3.2.2 Approach

Using the system descriptions of Subsection 15A.3.1, the event and fault trees (Figures 15A.1-2, 3 and 4 and 15A.3-2 and 15A.3-3) were constructed to show the failure paths that could result in high demand on all feedwater pumps.

15A.3.3 Results

The frequency of a Feedwater Controller Failure- Maximum Demand is found to be very low. The event should be treated as an Infrequent Event. The Event Tree is shown on Figure 15A.3-3 and the results of the analysis are shown on Table 15A-2.

15A.4 LOSS OF FEEDWATER HEATING WITH FAILURE OF SELECTED CONTROL ROD RUN-IN

A feedwater heating can be reduced because of

- unintended opening of a heater bypass line, or
- unintended closure of a steam extraction line to one of the feedwater heaters

The feedwater control system (FWCS) includes logic to reduce the consequences of a reduction in feedwater heating. Feedwater temperature is continuously monitored and when that temperature exceeds a ΔT setpoint then the FWCS sends an alarm to the operator and signals the RC&IS to initiate a selected control rods run-in (SCCRI) to reduce power and avoid a scram. However, the safety analysis for this event is analyzed at bounding ΔT rather than at the ΔT setpoint and without credit for event mitigation by an SCCRI.

There is no single failure that can lead to this condition and this event should be classified as an Infrequent Occurrence. The following discussion provides the basis for that classification.

15A.4.1 System Description

15A.4.1.1 Feedwater Heaters

Low pressure feedwater heating consists of three strings with four stages each. Two stages of each low pressure string are located within the condenser neck and do not have steam extraction isolation valves. The fifth stage of low pressure feedwater heating is provided by one direct contact feedwater heater tank. Each of the three LP condensers provides steam to the tank.

The high pressure heater strings consist of two strings with two stages each.

The low pressure and high pressure strings have a single feedwater heater bypass line containing a normally closed, motor-operated valve.

An erroneous high-level indication from 2/3 level transmitters will initiate control actions that open the bypass line around either the low pressure or high pressure heaters.

The number of heaters and valves that can experience a level transmitter failure is shown in Table 15A-4.

Figure 15A4-3 shows a simplified flow diagram of feedwater heating.

15A.4.1.2 Selected Control Rods Run-In (SCCRI)

The RC&IS utilizes a dual redundant architecture with an expected availability better or equal to 0.99. The expected availability is to be based on verified availability calculations (i.e., based on accepted standard reliability calculation methods).

The expected frequency of an inadvertent movement, of more than one control rod, due to failures in the RC&IS, is intended to be less than or equal to once in 100 reactor operating years. This expected frequency is based on verified reliability calculations.

Upon receipt of appropriate SCRRI initiation signals the RC&IS controls the FMCRD motors rods and drives the rods into their SCRRI target position. The SCRRI initiation signals are provided to RC&IS from the Feedwater Control System. Manual SCRRI initiation signals originate from the RC&IS Dedicated Operator Interface. Refer to Figure 15A.4-4 for a simplified depiction of the digital RCIS.

No single failure in the RC&IS results in failure to insert more than one operable control rod when the SCRRI function is activated.

Two manual operator actions shall be required for manual initiation of the SCRRI function.

Feedwater Control System: Each channel of the Feedwater Control System provides two channels of RC&IS with the SCRRI signals on loss of feed water heating. Validated total feedwater flow and feedwater temperature signals are provided to the RC&IS.

Upon receipt of SCRRI signal, RC&IS inserts a selected group of control rods to their SCRRI target positions.

The feedwater control system provides three channels indicating an SCRRI request signal. This request signal is generated when two out of three channels indicate a loss of feedwater heating condition.

15A.4.2 Analysis

The assumption for this analysis is that feedwater heater level signal failures from two channels occurs. One or more of these failures is not detected by the diagnostics and results in opening of either the HP or LP heater bypass valve.

At the same time, there is a failure of two out of three channels in the FWCS/RC&IS to generate the SCRRI signal. This signal would automatically drive in selected control rods and initiate a control room alarm, allowing the operator to terminate the event.

15A.4.2.1 Analytical Conditions

It was assumed that the failure rates represent failures that result in loss of output.

Mean-time-to-repair is shown on Table 15A-3, item 1.

Failure rates for electronic modules are estimated, based on anticipated complexity of the circuit functions as shown on Table 15A-3, item 2.

Spurious opening of a bypass valve or closing of a steam extraction valve was also evaluated.

15A.4.2.2 Approach

The event and fault trees (Figures 15A.1-2 through 15A.1-4 and 15A.4-1) were constructed to show the failure paths that could result in a loss of feedwater heating without SCCRI. The trees were developed using the system design information described in Subsection 15A.4.1.

The undetected failure of an SCRRI channel was modeled with a conservative assumption that surveillance frequencies were 1 year. The second channel failure was conservatively assumed to occur within the first 10 hours after a loss of feedwater heating event.

The failure probability from one FWH initiating the transient was then combined (one minus the product of the complements for the "OR'ed events) to give the cumulative probability for the event.

15A.4.3 Results

The frequency of a Loss of Feedwater Heating with Failure of SCRRI is found to be very low. The event should be treated as an Infrequent Event. The event frequency is shown on Figure 15A.4-2 and Table 15A-2.

15A.5 INADVERTENT SHUTDOWN COOLING FUNCTION OPERATION

Misoperation of the cooling water controls for the RWCU/SDC heat exchangers can lead to a moderate temperature decrease in the primary coolant system. Increased cooling is caused by malfunctioning of the cooling water controls for the RWCU/SDC heat exchangers. This event is analyzed at both power operation and during reactor startup. Scram may occur during startup.

There is no single failure that can lead to this condition. The probability of occurrence is low and this event should be classified as an Infrequent Event. The following discussion provides the basis for that classification.

15A.5.1 System Description

15A.5.1.1 RCCW

RCCW cooling water is supplied to two cooling water trains during normal operation with one pump in service and the other two pumps in standby. One heat exchanger functions during normal operation with two heat exchangers in standby. The cross-tie valves will be open during normal operation to allow operating flexibility. Major users of RCCW water (CWS, DCS, FAPCS, and RWCU/SDC) have redundant heat exchangers on each train, and only one of these trains will be in use during normal operating conditions. The train not in use will have the cooling water supply automatically isolated. This will reduce wear on the unused heat exchanger, and reduce the required cooling water flow rate during normal operation.

RCCW provides cooling water to the RWCU non-regenerative heat exchangers. During power operation, only one RWCU train will be in operation. The train not in use has RCCW automatically isolated. During startup or cooldown, both trains of RWCU/SDC will reject heat to RCCW.

The RCCW outlet temperature control valves for the RWCU/SDC heat exchangers are controlled by monitoring the temperature of the RWCU/SDC process fluid.

During normal power generation one pump is in operation, and the remaining two will be in standby. The standby pump(s) will start automatically on detection of either of the following:

- The operating pump trips
- Loss of Preferred Power initiated shutdown

15A.5.1.2 RWCU/SDC

The reactor water clean-up/shutdown cooling system consists of two independent and redundant trains. Each train has one pump, with an adjustable speed drive (ASD), one regenerative heat exchanger (RHX), one non-regenerative heat exchanger (NRHX), and one demineralizer unit along with valves, piping, strainers, controls and power inputs.

Each train of the system is sized to process the water in the primary system at a rate of 1% of rated feedwater flow during all modes of operation (except heatup and startup modes) including normal power operation, cooldown and shutdown operation.

Both trains operate simultaneously for all modes of operation except for Reactor Water Cleanup (Mode 8), End of Refueling (Mode 14), Reactor Well Cooling (Mode 15), and Reactor Boltup (Mode 16) when only one train operates.

The RWCU System is designed to ensure the RCCWS temperature exiting the NRHX does not exceed a specified maximum temperature during all modes of operation. It is designed with pump speed controls to ensure the NRHX heat transfer rate does not cause the RCCWS cold leg temperature to exceed its design temperature by more than about 2F°.

The RWCU/SDC water returning from the demineralizer is reheated to within 25ºF of the feedwater temperature during normal reactor water cleanup operation to minimize thermal stresses at the feedwater line connection.

A combination of pump speed adjustment and demineralizer bypass valve position is utilized to balance the pressure drop across the demineralizer during Low Pressure Shutdown Cooling (Modes 11 and 12), Refueling (Modes 13 and 14), Reactor Well Cooling (Mode 15) and Reactor Boltup (Mode 16).

15A.5.2 Analysis

The analysis is performed assuming the plant is in at power operation with one RWCU train in operation. It is conservatively assumed that the second train of RWCU begins to operate. The probability analysis assumes continuous full power operation and that this assumption bounds the total failure probability for all modes of operation.

A failure of any of the following is considered sufficient to cause a temperature decrease in the primary system:

- RWCU pump speed control
- RCCW temperature control failures
- RCCW multiple valve failures that increase cooling flow to the RWCU heat exchangers (i.e. a combination of reduced heat exchanger bypass flow and improper control action of the heat exchanger throttling flow must occur to cause an overcooling).

Refer to Figure 15A-1 for a depiction of the RWCU system and its controls.

15A.5.2.1 Analytical Conditions

The failure rate for the temperature transmitter to fail to a maximum is taken as half the value from Table 15A-3, item 6. The failure rate that is shown is for a failure in either direction. The upscale failure rate is assumed to be half of the total failure rate.

The failure rate for the pump speed controller is estimated as the failure rate for a solid state relay to operate spuriously to the energized state shown on Table 15A-3, item 3.

Surveillance testing or calibration is performed once a year on the temperature elements and speed controller, so that the failure rate is the hourly failure rate times 6 months.

15A.5.2.2 Approach

The event tree for failure of the RWCU pump was developed from the failure rate data assuming that the running pump operated for a full year. The failure rate for the standby pump was calculated assuming that the pump is called upon to run on the last day of this one year period.

The failure contribution from RCCW failures was ignored because there is no effect on the RWCU heat exchangers unless the RWCU speed control fails as well.

Failure of the pump speed controller and temperature element is required. Otherwise the control system will compensate for the individual component failure and limit the temperature reduction.

Inadvertent operation during shutdown was not evaluated because it is less likely that operation at power. The Event Tree would be constructed identically, but the failure rate for the operating pump would be based on the operating time of a refueling outage rather than on a full year of operation.

15A.5.3 Results

The frequency of an Inadvertent SDC/RWCU operation is found to be low. The Event Tree for Inadvertent RWCU is shown on Figure 15A.5-1 and Table 15A-2.

15A.6 INADVERTENT PRIMARY SYSTEM DEPRESSURIZATION

15A.6.1 Inadvertent Opening Of A Safety/Relief Valve

Inadvertent opening of a SRV (IOSRV) is attributed to malfunction of the valve or an operator initiated opening. Logic initiated opening of a single SRV is not credible as is shown in the subsequent discussion. Failure of multiple channels is required to produce an open demand signal and this signal would call for more than one valve to open. Failure to reclose after a transient is also addressed in this section. The stuck-open failure of a single SRV (SOSRV) in the absence of an opening signal is addressed in Subsection 15A.6.3.

15A.6.1.1 System Description

The ADS is a part of the ECCS and operates to depressurize the reactor for the low pressure GDCS to be able to make up coolant to the reactor. The ADS is composed of the SRVs and depressurization valves (DPVs) and their associated instrumentation and controls. The ADS consists of the 18 SRVs and 8 DPVs.

The ADS may also be manually initiated from the main control room.

The SRVs provide three main protection functions:

- overpressure safety operation (the valves are actuated by inlet steam pressure to prevent nuclear system overpressurization);
- depressurization operation [the values are actuated by the Automatic Depressurization Subsystem (ADS) as part of the Emergency Core Cooling System (ECCS) for events involving breaks in the nuclear system process barrier]; and
- overpressure relief operation (the valves are opened using a pneumatic actuator upon receipt of a manually-initiated signal to reduce pressure or to limit a pressure rise). Refer to Figure 15A.6-15.

The SRVs can be actuated in two ways: (1) by pneumatic actuation with electrical control power or (2) by mechanical actuation using inlet steam pressure at a predetermined setpoint. The suppression pool provides a heat sink for steam relieved by these valves.

Ten of the SRVs are designed as a dual function direct-acting type valve. Eight of the SRVs are designed as a direct-acting type valve. Each of the dual function SRVs is equipped with three solenoid-operated pilot valves powered by 250 VDC. Two of the solenoid valves shall each receive signals from the Class 1E SSLC system logic and the third solenoid valve shall receive signals from the DPS logic (Refer to Figure 15A.6-16). Pressurizing ADS Division I control logic actuates a solenoid pilot valve on each ADS SRV and ADS Division II control logic actuates a second separate solenoid pilot valve on each ADS SRV. Refer to Figure 15A.6-3 for the SRV actuation logic.

Any of the pilot valves can operate its associated ADS valve. The actuation of the solenoid-pilot valve causes the ADS SRV to open to provide depressurization.

These pilot valves control the pneumatic pressure applied by accumulators with N2. The operator can also control the SRV manually. Separate accumulators are provided with the control equipment to store pneumatic energy for relief valve operation.

The ADS initiating logic is executed in a two-out-of-four coincidence logic.

Sensors provide inputs to local multiplexer units that perform signal conditioning and analog-to-digital conversion. [lcl272]All four transmitter signals are fed into the two-out-of-four logic for each of the two divisions. If any two of the input signals agree there is a valid initiation signal, then the initial ADS timer is started. There are four separate and independent logic divisions (1, 2, 3 and 4) of ADS, with each division having both microprocessor-based logic units and non-microprocessor-based logic units. The SRV initiating logic is executed in the non-microprocessor-based logic units, while the DPV initiating logic is executed in the microprocessor-based logic units.

The system automatically defaults to 2-out-of-3 coincident voting in three operational divisions when a division of sensor inputs or trip logic is bypassed.

Electrical elements in the control system energize causing the relief valves to open. After receipt of the initiation signals and after a delay provided by time delay elements the solenoid pilot gas valves is energized. This allows pneumatic pressure from the accumulator to act on the gas cylinder operator. The gas cylinder operator opens and holds the ADS SRV open. Lights in the MCR indicate when the solenoid-operated pilot valves are energized to open an SRV Monitoring device mounted on the valve operators provides inputs for valve position to the performance monitoring and control subsystem of the plant computer system and the annunciators.

Manual actuation capability is provided to allow the operator to initiate ADS immediately (no time delay) if required. Such initiation is performed by first arming the ADS initiation switch for each of two channels within one of the two divisions.

15A.6.1.2 Analysis

The ESBWR uses a dual function, direct-acting type valve as opposed to a pilot-operated valve.

Therefore, the typical mechanisms that cause the pilot valve to open spuriously and to fail to open properly are not applicable to the ESBWR design. Historically, valve performance issues have been premature opening, delayed closing and sticking open. It is these mechanisms that have caused the most serious concerns with sticking and premature opening of safety/relief valves. By adopting a direct acting safety/relief valve design, these most serious concerns are eliminated in the ESBWR.

The evaluation of inadvertent opening therefore focuses on the following failure mechanisms.

- Inadvertent opening caused by the initiating logic. Note that the historical failure rate data (see Table 15A-3, items 4 and 17) does not reflect operating history with digital logic.
- Operator error, which would require multiple steps and would be correctable because the valve could be re-closed.
- Premature opening of the valve and delayed closing after a valid pressure increase.

15A.6.1.2.1 Analytical Conditions

Inadvertent Opening Caused by Initiation Logic

The assumption for this analysis is that ADV demand output signal failures from two channels has occurred, one or more of these failures has not been detected by the diagnostics and results in opening of one or more SRVs.

It was assumed that the failure rates represent an open demand on the SRV. Mean-time-to-repair is shown on Table 15A-3, item 1.

Failure rates for electronic modules are estimated, based on anticipated complexity of the circuit functions on Table 15A-3, item 2.

Improper Operation After A Pressure Increase

Data for BWR initiating events between the years 1988 and 2003 caused by a stuck open SRV is reported as shown on Table 15A-3, item 4. Refer to [lcl276]Table 15A-5 for a partial list of these events recorded between 1988 and 1995. As shown on the table, six of the ten failures were stuck open valves. The remaining failures were delayed closing of the valves. Updated data to 2003 shows a mean frequency as shown on Table 15A-3 item 4.

15A.6.1.2.2 Approach

The failure rate due to initiation logic was estimated using the system block diagram of Figures 15A.6-1 and 15A.6-2. The fault trees (Figures 15A.6-5 and 15A.6-9) were constructed to show the failure paths that could result in sensor failure.

The logic equations (Table 15A-1) were written from the event trees and after simplification were evaluated for the frequencies identified in the introduction.

The evaluation of the initiation logic shows that failures from within the SSLC logic are very low. Because the failure probability was so small, the value was not included in the overall failure rate for the DPS initiation system. An additional estimate was made for all other causes outside of this logic and for operator error causing a single valve to go open. For this analysis, it is conservatively assumed that the ESBWR design will achieve a reduction in the frequency of valves failing to reseat by a factor of 3 over the failure rate for pilot-operated SRVs. Main steam safety/relief valves are similar to the dual-function direct-acting SRV types currently in service in GE BWR/5 and BWR/6 plants. These SRV types have demonstrated improved in-service performance and reliability as compared to pilot-operated safety/relief valves used on earlier BWR models.[lcl280]

Failures at the actuated device are not treated in this evaluation. Rather they are included in the evaluation of stuck open SRVs because these failures will cause the valve to remain open rather then to open and re-close.

Operator error was conservatively estimated as 10% of all the historical ISORV events. None of the failures listed on Table 15A-5 were caused by an operator inadvertently opening an SRV. The 10% estimate is also conservative because it assumes that the error results in an open SRV that subsequently sticks open.

15A.6.1.3 Results

The failure rate for a single SRV is estimated from a spurious logic error calling for valve opening and from a failure at the valve actuation circuit. Inadvertent initiation signals originating from the dual redundant SSLC/E-DCIS are very small, as shown on Figure 15A.6-8. The overall frequency of Inadvertent Opening of an SRV is shown on Figure 15A.6-9 and Table 15A-2. These results are consistent with an improvement in valve failure rates by about a factor of two. The event should be treated as an Infrequent Event

15A.6.2 Inadvertent Opening Of A Depressurization Valve

15A.6.2.1 System Description

The auto-depressurization subsystem consists of a set of conventional safety-relief type valves (SRVs) and associated discharge lines that are terminated with quenchers located in the suppression pool. In addition, the auto-depressurization subsystem consists of a back-up set of squib-type pyrotechnic-actuated depressurization valves (DPVs) that when opened achieve reactor depressurization by discharging steam directly into the drywell. The ADS consists of the

18 SRVs (10 of which are dual function valves) and eight DPVs. The ADS is designed to minimize the potential for interruption of normal plant operation as a result of excessive component leakage or inadvertent actuation without diminishing the safety of the system. The DPV valves are squib-actuated non-reclosing valves with a metal diaphragm seal and are sealed with an end cap. Refer to Figure 15A.6-15. The seal cap is sheared off the valve by an explosive initiator-booster. The end cap prevents leakage during normal operation. The valves are about twice the size of an SRV. Two initiator boosters (squibs), singly or jointly actuate the shearing plunger. The squib is actuated by either of two independent firing circuits.

The ADS initiation logic is described in more detail in Subsection 15A.6.2.1.2. Figure 15A.6-4 shows the DPV local actuation logic. [lcl282]Reliability testing was performed on the valves to both fire and to avoid accidental firing. The tests included irradiation thermal aging and LOCA environmental conditions.

The Diverse Protection System provides an independent, separate and diverse means of actuating the SRVs and DPVs. The Diverse Protection System provides acquisition of data from NBS sensors, trip determination and trip output signals for actuating the SRVs.

ESF initiation logic from SRV, and DPV are included in the Diverse Protection System (DPS) that provides diverse emergency core cooling system (ECCS) protections. The DPS provides diverse ESF logic for the SRV solenoid-controlled valves opening at low water level (L1) and the DPV squib-initiation valves opening at low water level (L1).

This set of diverse ESF logics resides in separate and independent hardware and software equipment from the primary ESF systems. The process variables sensors that provide inputs to this diverse set of logics use different sets of sensors from that used in the primary ESF systems. The diverse logic equipment is non-safety-related with triplicate redundant channels. The diverse equipment power source is non-safety-related. The initiation logic is "energize to actuate" similar to the primary ESF. The trip logic is based on 2-out-of-3 voting.

For the SRV opening function, two of the three SRV solenoids on each SRV are powered by two of the four divisional Class 1E power sources in the primary ESF ADS system. A third solenoid on each SRV is powered by the non-safety-related load group, with the trip logic controlled by the diverse DPS. All ten SRVs in the ADS are controlled by the DPS through the third solenoid on each valve.

For the DPV opening function, one of the two squib initiators on each DPV is controlled by and connected to the non-safety-related DPS logic. However, the two squib initiators on each of all eight DPVs are all controlled simultaneously by the primary ESF ADS logic. The reliability and availability of DPV initiation by the primary ESF ADS function is not affected by the DPS logic. The logic contacts circuit from the DPS is arranged in parallel with the ESF circuit. It takes two simultaneous SSLC/ESF trip outputs to initiate the DPV squib valve opening. It also takes two simultaneous DPS trip outputs to initiate the DPV squib valve opening.

15A.6.2.2 Analysis

The assumption for this analysis is that spurious opening of a single DPV occurs from one of three sources:

- Spurious demand signal.

  - ADS demand output signal failures from two channels have occurred, and one or more of these failures has not been detected by the diagnostics results in opening of one DPV.
  - common cause failure of load drivers results in a single spurious opening of a DPV.
- Local failure mechanisms occur at the valve igniter.
- Operator error.

where 0.1 is the generic common cause failure factor assumed for spurious operation. As developed below, the DPV spurious opening rate to be used is 4.4 x 10^{-4} events/year.

15A.6.2.2.1 Analytical Conditions

[lcl285]The failure rate for the spurious opening of an explosive valve is shown on Table 15A-3, item 5.

Inadvertent initiation signals originating from the dual redundant SSLC/E-DCIS are very small as shown on Figure 15A.6-10. This failure mechanism is ignored because a failure at this level would initiate opening of more than one DPV.

Figure 15A.6-4 shows the DPV valve initiation logic.

Mean-time-to-repair is shown on Table 15A-3, item 1.

Spurious demand signal

Failure rate for a solid state relay to operate spuriously to the energized state is shown on Table 15A-3, item 3.

Common cause failure of load drivers could result in a single spurious opening of a DPV. The frequency of a CCF for load driver is:

$2.0 \times 10^{-7} \times 8760 \times 0.1 = 4.4 \times 10^{-4}$ events/year

where 0.1 is the generic common cause failure factor assumed for spurious operation.

Local failure mechanisms occur at the valve igniter. [lcl287]

The failure rate for the spurious opening of an explosive valve is shown on Table 15A-3, item 5. This failure rate was reduced by a factor of 4 for the following reasons:

- Testing of the specific valve design under radiation, temperature, humidity and pressure conditions without spurious activation. Testing of the devices for automotive use is performed at 107^0C for 400 hours. Overtest at 107^0C for 800 hours have also shown no effect.
- Use of igniters that have been used in the automotive industry as air-bag igniters since 1987. Between 1987 and 1993 (the period for which data are available) there were no reported failures in the more than 15,000,000 automotive igniters that were delivered.
- The propellant used in the DPV is more stable than the igniter.

Operator error.

An additional failure mechanism resulting from an operator error was also included in the model. This estimate is bounding and conservative because:

- The ESBWR design reduces the risk of operator error during surveillance testing.
- The DPV valve is not tested while installed in the plant. Only the initiation circuitry is subject to surveillance testing.
- The opening of the DPV is administratively controlled by key lock.

15A.6.2.2.2 Approach

The failure rate due to initiation logic was developed using the system block diagram of Figures 15A.6-1, 15A.6-2. The same event and fault trees that were used for the IOSRV (Figures 15A.6-5 and 15A.6-9) were used to show the failure paths that could result in sensor failure.

The failure rate for the ADS solid state relay that energizes the igniter and spurious explosive valve operation are also modeled in the overall failure rate as shown on Figure 15A.6-11.

The failure rate of the explosive valve has been calculated for the ESBWR PRA. The value is listed on Table 15A-3, item 19.

15A.6.2.3 Results

The Event Tree for Inadvertent Opening of a DPV is shown on Figure 15A.6-12 and the estimated failure rate is shown on Table 15A-2. Based on the low failure probability the, IDPVO is classified as an Infrequent Event.

15A.6.3 Stuck Open Safety/Relief Valve

A stuck open SRV (SOSRV) can result from either a valid opening or inadvertent opening of the SRVs with failure of one valve to close. This event should be classified as an Infrequent Event as discussed in the following discussion.

15A.6.3.1 System Description

Subsection 15A.6.1.1 describes the Safety Relief Valves and the Safety System Logic Control System that automatically initiates opening of the SRVs. Subsection 15A.6.2.1 describes the Diverse Protection System logic.

15A.6.3.2 Analysis

The assumption for this analysis is that a stuck open SRV can only occur because of local failure mechanisms or operator error. Local failures occur because of mechanical failure of the valve or multiple failures of the solid state relays (loaders) shown on Figure 15A.6-3.

Initiation logic errors would not open a single valve and are not considered in the evaluation.

15A.6.3.2.1 Analytical Conditions

Valve Mechanical Failure

Table 15A-5 provides a list of Inadvertent Open and Stuck SRVs recorded between 1988 and 1995. Table 15A-3, item 17 shows the mean frequency for these events. As shown on the table, six of the ten failures were stuck open valves. The remaining failures were delayed closing of the valves. Updated data to 2003 have a mean frequency as shown on Table 15A-3, item 4 based on 13 failures in 441.6 years of BWR operation at critically.

The failure rate for the spurious opening of an SRV is shown on Table 15A-3, item 4. Note that this value applies to the mechanical failure of pilot operated SRVs. The SRVs used in the ESBWR do not have a pilot stage. The ESBWR uses a direct-acting SRV design. Therefore, the mechanisms that cause the pilot valve to fail to open properly have been eliminated from the design. This design has operated successfully in the BWR 5 and BWR 6. The failure rate data in table for the pilot-operated valves has been reduced by a factor of three in this evaluation, which is conservatively low given the operating history of the direct acting valves.

Solid State Relay Failure

The failure rate for the ADS solid state relay that energizes the SRV solenoid are also modeled in the overall failure rate as shown on Figure 15A.6-13. Failure rate for a solid state relay to operate spuriously to the energized state is shown on Table 15A-3, item 3.

Figure 15A.6-3 shows the SRV valve initiation logic.

Operator Error

The contribution of SOSRV from operator error is conservatively estimated to be 10%. Refer to Subsection 15A.6.1.3.2.

15A.6.3.2.2 Approach

The failure rate for a single SRV is estimated from a spurious logic error calling for valve opening and from a failure at the valve actuation circuit. Inadvertent initiation signals originating from the dual redundant SSLC/E-DCIS are very small as shown on Figure 15A.6-8.

Mechanical failure of the ESBWR SRVs is much lower than the historical data for BWRs for the following reasons:

- Main steam safety/relief valves for the ESBWR service are similar to the dual-function direct-acting SRV types currently in service in GE BWR/5 and BWR/6 plants. These SRV types have demonstrated improved in-service performance and reliability as compared to pilot-operated safety/relief valves used on earlier BWR models.
- The typical mechanisms that cause the pilot valve to open spuriously and to fail to open properly are not applicable to the Lungmen NPS design. It is these mechanisms that have caused the most serious concerns with the Target Rock safety/relief valve performance. By adopting a direct acting safety/relief valve design, these most serious concerns are eliminated in the ESBWR.
- The ESBWR does not have Infrequent Events that open the SRVs on high pressure.

Operator error is not a likely cause of single valve failure because:

- The ESBWR design reduces the risk of operator error during surveillance testing.
- The opening of the SRV is administratively controlled by key lock.

15A.6.3.3 Results

The Event Tree for a Stuck Open SRV is shown on Figure 15A.6-14. The overall frequency for a Stuck Open Safety Relief Valve is shown on Table 15A-2.

15A.7 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

15A.7.1 Control Rod Withdrawal Error (RWE)

The causes of a potential rod withdrawal error (RWE) transient are either a procedural error by the operator in which a single control rod or a gang of control rods is withdrawn continuously, or a malfunction of the automated rod withdrawal sequence control logic during automated operation in which a gang of control rods is withdrawn continuously. In either case, the operating thermal limits rod block function would block any further rod withdrawal when the operating thermal limit is reached. That is, the withdrawal of rods would be stopped before the operating thermal limit is reached. Because there is no operating limit violation due to the preventive function of the automatic thermal limit monitor (ATLM), there is no rod withdrawal error transient event.

ATLM performs the rod block monitoring function. The ATLM is a dual channel subsystem of the rod control and information system (RCIS). In each ATLM channel there are two independent thermal limit monitoring devices. One device monitors the MCPR limit and protects the operating limit MCPR, and the other device monitors the APLHGR limit and protects the operating limit of the APLHGR. The rod block algorithm and setpoint of the ATLM are based on actual on-line core thermal limit information. If any one of the two limits is reached, either due to control rod withdrawal or recirculation flow increase, the control rod withdrawal permissive is removed.

For a RWE event to occur, both channels of ATLM have to fail at the same time when the control rod is in motion.

15A.7.1.1 Control Rod Withdrawal Error During Refueling

The event considered here is inadvertent criticality due to the complete withdrawal or removal of the most reactive rod during refueling. The probability of the initial causes, alone, is considered low enough to warrant its being categorized as an Infrequent Event, because there is no postulated set of circumstances that results in an inadvertent RWE while in the REFUEL mode.

15A.7.1.1.1 System Description

During refueling operations, system interlocks provide assurance that inadvertent criticality does not occur because of a control rod withdrawal.

15A.7.1.1.2 Analysis

Fuel Insertion with Control Rod Withdrawn

All control rods are fully inserted when fuel is being loaded into the core to minimize the possibility of loading fuel into a cell containing no control rod. This requirement is backed up by refueling interlocks during both rod withdrawal and movement of the refueling platform. When the mode switch is in the REFUEL position, the interlocks prevent the platform from being moved over the core if a control rod is withdrawn and fuel is on the hoist. Likewise, if the refueling platform is over the core and fuel is on the hoist, control rod motion is blocked by the interlocks.

Second Control Rod Removal or Withdrawal

When the platform is not over the core (or fuel is not on the hoist) and the mode switch is in the REFUEL position, only one control rod can be withdrawn when the RC&IS SINGLE/GANG switch is in the SINGLE position. When the RC&IS switch is in the GANG position, only one control rod pair with the same HCU may be withdrawn. Any attempt to withdraw an additional rod results in a rod block by the RC&IS interlock. Because the core is designed to meet shutdown requirements with one control rod pair (with the same HCU) or one rod of maximum worth withdrawn, the core remains subcritical even with one rod withdrawn.

Control Rod Removal Without Fuel Removal

The design of the control rod incorporates a bayonet coupling system that physically prevents the upward removal of the control rod without:

- decoupling by rotation, and
- the simultaneous or prior removal of the four adjacent fuel bundles.

15A.7.1.1.2.1 Analytical Conditions

It was assumed that the failure rates represent failures that result in failure to generate a Rod Withdrawal Inhibit.

Failure rates for electronic modules are estimated, based on anticipated complexity of the circuit functions as shown on Table 15A-3, item 2.

The Event Tree makes the following conservative assumptions:

- The frequency of a rod withdrawal error is once per year.
- The failure of either channel can occur any time during the year.
- The probability of detecting the second failure is 1.

15A.7.1.1.2.2 Approach

The event tree of Figure 15A.7-1 was constructed to show the failure paths that could result in failure of a Rod Withdrawal Inhibit signal while a rod withdrawal was occurring. The only credible path is by the undetected failure of both the channels.

15A.7.1.1.3 Results

The Event Tree for a Rod Withdrawal error During Refueling is shown on Figure 15A.7-1. The results of the failure analysis are shown on Table 15A-2. The frequency of a Rod Withdrawal Error during Refueling with a failure of the Rod Withdrawal inhibit function is extremely low. The event can be treated as an Infrequent Event.

15A.7.1.2 Control Rod Withdrawal Error During Startup

It is postulated that during a reactor startup, a gang of control rods or a single control rod is inadvertently withdrawn continuously due to a procedural error by the operator following a double failure in the RC&IS or a double failure of the Plant Automation System.

15A.7.1.2.1 System Description

During startup, system interlocks provide assurance that inadvertent criticality does not occur because of a control rod withdrawal.

15A.7.1.2.2 Analysis

The RC&IS has a dual channel rod worth minimizer function that prevents withdrawal of any out-of-sequence rods from 100% to 50% control rod density, i.e., for Group 1 to Group 4 rods. It also has bank position withdraw sequence constraints such that, if the withdraw sequence constraints are violated, the rod worth minimizer function of the RC&IS initiates a rod block. The bank position constraints are in effect from 50% control rod density to the low power setpoint.

The Plant Automation System consists of a redundant, triplicated process controller. It provides demand signals to the RC&IS to position the control rods during an automatic startup. The Plant Automation System is described in Subsection 7.7.4.

In addition, the startup range neutron monitors (SRNM), a subsystem of the Neutron Monitoring System (NMS), has a "period withdrawal permissive" rod block interlock on each of eight SRNM channels. When any of these eight SRNM channels senses that the reactor period reaches the rod block setpoint due to erroneous control rod withdrawal, control rod withdrawal is blocked. As a result, continuous control rod withdrawal is stopped. The rod block setpoint is so selected that no reactor scram, nor violation of thermal margins, occurs during this event.

Because both the RC&IS and Plant Automation System consist of either a dual channel or triplicated processors, no single failure can cause this event to occur. In accordance with Standard Review Plan 15.4.1, this event need not be considered.

15A.7.1.2.2.1 Analytical Conditions

It was assumed that the failure rates represent failures that result in failure to generate a Rod Withdrawal Inhibit.

Failure rates for electronic modules are estimated, based on anticipated complexity of the circuit functions as shown on Table 15A-3, item 2.

The Event Tree makes the following conservative assumptions:

- the frequency of a rod withdrawal error is once per year.

- the failure of either channel can occur any time during the year.
- the probability of detecting the second failure is 1.

15A.7.1.2.2.2 Approach

The event tree of Figure 15A.7-1 was constructed to show the failure paths that could result in failure of a Rod Withdrawal Inhibit signal while a rod withdrawal was occurring. The only credible path is by the undetected failure of both the channels.

15A.7.1.2.3 Results

The Event Tree for a Rod Withdrawal error During Startup is shown on Figure 15A.7-1. The results of the failure analysis are shown on Table 15A-2. The frequency of a Rod Withdrawal Error during Startup with a failure of the Rod Withdrawal inhibit function is extremely low. The event can be treated as an Infrequent Event.

15A.7.1.3 Control Rod Withdrawal Error During Power Operation

The causes of a potential RWE are either a procedural error by the operator in which a single control rod or a gang of control rods is withdrawn continuously, or a malfunction of the automated rod withdrawal sequence control logic during automated operation in which a gang of control rods is withdrawn continuously. But in either case, the operating thermal limits rod block function block any further rod withdrawal when the operating thermal limit is reached. That is, the withdrawal of rods is stopped before the operating thermal limit is reached. The performance of the ATLM subsystem of the RC&IS prevents the RWE event from occurring. The core and system performance are not affected by such an operator error or control logic malfunction. There is no need to analyze this event.

15A.7.1.3.1 System Description

In the ESBWR, the automated thermal limit monitor (ATLM) subsystem performs the rod block monitoring function. The ATLM is a dual channel subsystem of the RC&IS. Each ATLM channel has three independent thermal limit monitoring functions. One function monitors the MCPR limit and protects the operating limit MCPR, another function also monitors the MCPR limit and protects the safety limit MCPR, and the third function monitors the APLHGR limit and protects the operating limit of the APLHGR. The rod block algorithm and setpoint of the ATLM are based on actual on-line core thermal limit information. If any one of the three limits is reached, such as due to control rod withdrawal, control rod withdrawal permissive is removed.

15A.7.1.3.2 Analysis

15A.7.1.3.2.1 Analytical Conditions

It was assumed that the failure rates represent failures that result in failure to generate a Rod Withdrawal Inhibit. Failure rates for electronic modules are estimated, based on anticipated complexity of the circuit functions as shown on Table 15A-3, item 2.

The Event Tree makes the following conservative assumptions:

- The frequency of a rod withdrawal error is once per year.

- The failure of either channel can occur any time during the year.
- The probability of detecting the second failure is 1.

15A.7.1.3.2.2 Approach

The event tree of Figure 15A.7-1 was constructed to show the failure paths that could result in failure of a Rod Withdrawal Inhibit signal while a rod withdrawal was occurring. The only credible path is by the undetected failure of both the channels.

15A.7.1.3.3 Results

The Event Tree for a Rod Withdrawal error Power operation is shown on Figure 15A.7-1. The results of the failure analysis are shown on Table 15A-2. The frequency of a Rod Withdrawal Error during Power Operation with a failure of the Rod Withdrawal inhibit function is extremely low. The event can be treated as an Infrequent Event.

15A.7.1.4 Fuel Assembly Loading Error, Mislocated Bundle

The loading of a fuel bundle in an improper location with subsequent operation of the core requires three separate and independent errors.

15A.7.1.4.1 System Description

The loading of a fuel bundle in an improper location with subsequent operation of the core requires three separate and independent errors:

- A bundle must be placed into a wrong location in the core.
- The bundle that was supposed to be loaded where the mislocation occurred is also put in an incorrect location or discharged.
- The misplaced bundles are overlooked during the core verification process performed following core loading.

Provisions to prevent potential fuel loading errors are included in the plant Operating Procedures /Technical Specifications.

15A.7.1.4.2 Analysis

Proper location of the fuel assembly in the reactor core is readily verified by visual observation and assured by verification procedures during core loading. GE provides recommended fuel assembly loading instructions for the initial core as part of the Startup Test Instructions (STIs). It is expected that the plant owners use similar procedures during subsequent refueling operations. Verification procedures include inventory checks, current bundle location logs, serial number verifications and visual or photographic inspection of the loaded core. The verification procedures are designed to minimize the possibility of the occurrence of the mislocated bundle accident.

15A.7.1.4.2.1 Analytical Conditions

The following fuel loading errors are possible in an initial core:

- A high–enrichment bundle is misloaded into a low–enrichment bundle location.

- A medium–enrichment bundle is misloaded into a low–enrichment bundle location.
- A low–enrichment bundle is misloaded into a high–enrichment bundle location.
- A low–enrichment bundle is misloaded into a medium–enrichment bundle location.
- A medium–enrichment bundle is misloaded into a high–enrichment bundle location.
- A high–enrichment bundle is misloaded into a medium–enrichment bundle location.

The only fuel bundle loading error that reduced thermal margin occurs when a high–enrichment bundle is interchanged with a medium–enrichment bundle located away from an LPRM.

For Reload Cores, the loading error involves the mislocation of at least two fuel bundles. One location is loaded with a bundle that would otherwise operate at a lower critical power. The other location would operate at a higher critical power. The low critical power location could have less margin to boiling transition than other bundles in the core; therefore, the MCPR operating limit is set to protect against this occurrence. There is a strong possibility that the core monitor will recognize and mitigate the consequences of a mislocated bundle. In the situation where the high radial power mislocated bundle is adjacent to an instrument, the power monitoring systems will indicate a higher monitored bundle power. The reactor will be operated with the most limiting of the bundles near the mislocation below the operating limit MCPR. A less advantageous situation is where the mislocated bundle has a bundle between it and an instrument. An ineffective situation occurs when the core monitor will not recognize the mislocation if the monitoring system is not radially TIP or LPRM adapted.

15A.7.1.4.2.2 Approach

The likelihood of operating the core with a mislocated bundle is low because multiple errors are required. The likelihood of a mislocation resulting in a reduced thermal margin is also low. In an initial core most mislocations do not cause adverse effects on thermal margin. For reload cores, at least two bundles have to be mislocated and fuel locations are verified. Verification procedures include inventory checks, current bundle location logs, serial number verifications and visual or photographic inspection of the loaded core.

15A.7.1.4.3 Results

The frequency of a Mislocated Fuel Assembly during Power Operation is low. The event can be treated as an Infrequent Event.

15A.7.1.5 Fuel Assembly Loading Error, Misoriented Bundle

The loading of a fuel bundle with an improper orientation with subsequent operation of the core requires includes five separate verifications.

15A.7.1.5.1 System Description

Not applicable

15A.7.1.5.2 Analysis

Proper orientation of fuel assemblies in the reactor core is readily verified by visual observation and assured by verification procedures during core loading.

Five separate visual indications of proper fuel assembly orientation exist:

- The channel fastener assemblies, including the spring and guard used to maintain clearances between channels, are located at one corner of each fuel assembly adjacent to the center of the control rod.
- The identification boss on the fuel assembly handle points toward the adjacent control rod.
- The channel spacing buttons are adjacent to the control rod passage area.
- The assembly identification numbers that are located on the fuel assembly handles are all readable from the direction of the center of the cell.
- There is cell–to–cell replication.

15A.7.1.5.2.1 Analytical Conditions

Not applicable

15A.7.1.5.2.2 Approach

Experience has demonstrated that these design features are clearly visible so that any misoriented fuel assembly would be readily identifiable during core loading verification. Verification procedures include inventory checks, current bundle location logs, serial number verifications and visual or photographic inspection of the loaded core.

15A.7.1.5.3 Results

The frequency of a Misoriented Fuel Assembly during Power Operation is low. The event can be treated as an Infrequent Event.

15A.8 RADIOACTIVITY RELEASE

15A.8.1 Waste Gas System Leak or Failure

The failure of a single active component leading to a direct release of radioactive gases to the environment is highly unlikely. Inadvertent operator action with bypass of the delay charcoal beds is analyzed for compliance to ESTB 11-5. The radiological analysis of this event in Chapter 15 evaluates the potential radiological consequences of an inadvertent bypass of the charcoal beds. It was assumed that operator error or computer error has led to the bypass of the eight follow-on beds in addition to the failure of the automated air-operated downstream isolation valve. Even with the failure of the downstream isolation valve, it is not anticipated or assumed that the isolation instrumentation would fail, but would instead alarm the control room with a high radiation alarm, causing the operator to manually isolate the Offgas System (i.e., close suction valves) within 30 minutes of the alarm.

15A.8.1.1 System Description

The ESBWR Offgas System is detailed in Section 11.3. To bypass either pathway (1) or (2) above requires the operator to enter a computer command with a required permissive. To bypass all tanks [pathway (3)] requires the operator to key in the command with two separate permissives. Because pathway (3) would require both inadvertent operation upon the operator (keying in the wrong command) plus getting two specific permissives for three incorrect decisions, it is not assumed that inadvertent entry into pathway (3) is likely to occur. Redundant upon human decision making and downstream of the charcoal beds shown on Figure 15.7-1 are a series of two redundant radiation monitoring instruments and an air-operated isolation valve. Upon receiving a Hi signal, the system alarms the control room, where a Hi-Hi signal will automatically initiate a "treat" mode, causing flow to process through the charcoal tanks, and override switch settings. Beyond this signal, a Hi-Hi-Hi signal will automatically shut off all flow from the Offgas System for radioactivity levels in excess of environmental limits, which are defined by 10CFR20 as not greater than 20 μSv/hr (2 mrem/hr) at the site boundary for any single hour or 7 μSv/hr (0.7 mRem/hr) over a seven-day period. Therefore, bypass of the charcoal beds during periods with significant radioactive flow through the Offgas System will be limited and/or automatically terminated by actuation of the downstream sensors.

15A.8.1.2 Analysis

15A.8.1.2.1 Analytical Conditions

Total U.S. reactor experience (1969–1997) 1,019 PWR critical years and 525 BWR critical years as reported in NUREG/CR-5750. This is a total of 1,544 critical years of operation through 1997. To date there has not been a direct release of the contents of a waste gas decay tank or other direct release to the environment.

15A.8.1.2.2 Approach

An event with a probability of occurrence of less than .01/year will be considered an Infrequent Occurrence. Given that there have been no events of this type in 1544 critical years reported in NUREG/CR-5750 (or since) the probability of occurrence can be considered to be less than .01/year. If this event did indeed have a probability of occurrence of .01/year or greater, then the probability that no such event has been experienced in over 1544 years of operating history is less than 10E-6.

15A.8.1.3 Results

The probability of occurrence of an uncontrolled direct release of offgas to the environment is less than .01/year. Therefore, this event should be classified as an Infrequent Event.

15A.8.2 Liquid-Containing Tank Failure

An unspecified event causes the complete release of the radioactive inventory in all tanks containing radionuclides in the liquid radwaste system. Postulated events that could cause a release of the inventory of a tank are sudden unmonitored cracks in the vessel or operator error. Small cracks and consequent low level releases are bounded by this analysis and should be contained without any significant release.

The ESBWR Radwaste Building is a UBC structure designed to withstand all credible seismic events. In addition, the walls of all compartments containing high-level liquid radwaste are lined up to a height capable of containing the release of all the liquid radwaste in the compartment. Because of these design capabilities, it is considered remote that any major accident involving the release of liquid radwaste into these volumes would result in the release of these liquids to the environment via the liquid pathway. Releases as a result of major cracks would instead result in the release of the liquid radwaste to the compartment and then to the building sump system for containment in other tanks or emergency tanks. A complete description of the liquid radwaste system is found in Section 11.2, except for the tank inventories, which are found in Section 12.2.

A liquid radwaste release caused by operator error is also considered a remote possibility. Operating techniques and administrative procedures emphasize detailed system and equipment operating instruction. A positive action interlock system is also provided to prevent inadvertent opening of a drain valve. Should a release of wastes occur, the lining would contain the release until the floor drain sump pumps in the building capture and contain such spills.

The probability of a complete tank release is considered low enough to warrant this event as a limiting fault.

15A.8.2.1 System Description

Not applicable

15A.8.2.2 Analysis

15A.8.2.2.1 Analytical Conditions

Total U.S. reactor experience (1969–1997) 1,019 PWR critical years and 525 BWR critical years as reported in NUREG/CR-5750. This is a total of 1,544 critical years of operation through 1997. To date there has not been a direct release of the contents of a waste gas decay tank or other direct release to the environment.

15A.8.2.2.2 Approach

An event with a probability of occurrence of less than .01/year will be considered an Infrequent Occurrence. Given that there have been no events of this type in 1544 critical years reported in NUREG/CR-5750 (or since) the probability of occurrence can be considered to be less than .01/year. If this event did indeed have a probability of occurrence of .01/year or greater, then the probability that no such event has been experienced in over 1544 years of operating history is less than 10E-6.

15A.8.2.3 Results

The probability of occurrence of an uncontrolled direct release of liquid waste to the environment is less than .01/year. Therefore, this event should be classified as an Infrequent Event.

15A.8.3 Spent Fuel Cask Drop Accident

Due to the redundant nature of the crane, the cask drop accident is not believed to be a credible accident.

15A.8.3.1 System Description

Not applicable

15A.8.3.2 Analysis

15A.8.3.2.1 Analytical Conditions

Total U.S. reactor experience (1969–1997) 1,019 PWR critical years and 525 BWR critical years as reported in NUREG/CR-5750. This is a total of 1,544 critical years of operation through 1997. To date there has not been a direct release of the contents of a waste gas decay tank or other direct release to the environment.

15A.8.3.2.2 Approach

An event with a probability of occurrence of less than .01/year will be considered an Infrequent Occurrence. Given that there have been no events of this type in 1544 critical years reported in NUREG/CR-5750 (or since) the probability of occurrence can be considered to be less than .01/year. If this event did indeed have a probability of occurrence of .01/year or greater, then the probability that no such event has been experienced in over 1544 years of operating history is less than 10E-6.

15A.8.3.3 Results

The probability of occurrence of a spent fuel cask drop accident is less than .01/year. Therefore, this event should be classified as an Infrequent Event.

Table 15A-1

Logic Equations

Failure probability is calculated as one minus the product of the complements, when the events are "OR" ed. For example, the failure probability for on of three components failing is calculated as:

$P = [1 – (1-B_1) (1-B_2) (1-B_3)]$

$= 1- (1- B_1 B_2 B_3 + B_1 B_2 + B_1 B_3 + B_2 B_3 - B_1 - B_2 – B_3)]$

$P = B_1 B_2 B_3 – (B_1 B_2 + B_1 B_3 + B_2 B_3) + (B_1 + B_2 + B_3)$

If $B_1 = B_2 = B_3 = B$, then

$P = B^3 - 3B^2 + 3B$

If $B << 1$, then

$P \simeq 3B$

This same procedure would be used for any number of components.

Table 15A-2

Summary of Event Frequency Estimates

Event	Estimated Frequency	Event Classification
Pressure Regulator Downscale Failure	6.9E-6/yr	Infrequent Event
Pressure Regulator Upscale Failure	6.9E-6/yr	Infrequent Event
Turbine Trip with Failure of All TBVs	.0028/yr	Infrequent Event
Generator Load Rejection with Failure of All TBVs	<<0.0047/yr	Infrequent Event
Feedwater Controller Failure - Maximum Demand	<<0.01/yr	Infrequent Event
Loss of Feedwater Heating With Failure of Selected Control Rod Run-In	0.00015/yr	Infrequent Event
Inadvertent Shutdown Cooling Function Operation	<<0.01	Infrequent Event
Inadvertent Opening Of A Safety/Relief Valve	0.0075/yr	Infrequent Event
Inadvertent Opening Of A Depressurization Valve	0.01 /yr	Infrequent Event
Stuck Open Safety/Relief Valve	0.007/yr	Infrequent Event
Control Rod Withdrawal Error During Refueling	<<1.0E-5/yr	Infrequent Event
Control Rod Withdrawal Error During Startup	<<1.0E-5/yr	Infrequent Event
Control Rod Withdrawal Error During Power Operation	<<1.0E-5/yr	Infrequent Event
Fuel Assembly Loading Error, Mislocated Bundle	<<0.01/yr	Infrequent Event
Fuel Assembly Loading Error, Mislocated Bundle	<<0.01/yr	Infrequent Event
Liquid-Containing Tank Failure	<<0.01/yr	Infrequent Event
Spent Fuel Cask Drop Accident	<<0.01/yr	Infrequent Event
Waste Gas System Leak or Failure	<<0.01/yr	Infrequent Event

Table 15A-3

Summary of Information Used in Failure Rate Estimates

Item No.	Item	Estimate	Reference
1	Mean time to repair (MTTR)	10 hours	ABWR Appendix 15D, Subsection 15D.3.1
2	Failure rates for electronic processors	1.0E-5/hr.	ABWR Appendix 15D, Subsection 15D.3.1
3	Failure rate for solid state relay operates: spuriously to energized state	2.0E-7/hr	ALWR Volume II, Chapter 1 Annex A, page A.A-28
4	Initiating events caused by stuck open SRV between 1988 and 2003	1.94E-2/yr.	NUREG/CR-5750. Initiating Events Study. 2003 Update, Page 2
5	Failure rate for spurious opening of an explosive valve	4.0E-7/hr.	ALWR Volume II, Chapter 1 Annex A, page A.A-25
6	Temperature transmitter fails to zero or maximum	4.0E-6/hr	ALWR Volume II, Chapter 1 Annex A, page A.A-29
7	Motor operated valve fails to remain open	1.4E-7/hr	ALWR Volume II, Chapter 1 Annex A, page A.A-25
8	Motor-Operated Valve fails to operate on demand	4.0E-3/day = 1.666E-4/hr	ALWR Volume II, Chapter 1 Annex A, page A.A-25
9	Air-operated valve transfers closed	1.5E-7/hr	ALWR Volume II, Chapter 1 Annex A, page A.A-25
10	Motor-driven pump fails to run	2.5E-5/hr	ALWR Volume II, Chapter 1 Annex A, page A.A-26
11	Motor-driven pump fails on demand	2.0 E-3/day	ALWR Volume II, Chapter 1 Annex A, page A.A-26

Table 15A-3

Summary of Information Used in Failure Rate Estimates

12	Initiating events caused by generator Load Rejection	0.015/yr	PRA Initiating Events Analysis 092-134-E-Z-0001, R0 Table 19.3.6-1
13	Initiating events caused by Turbine Trip	1.05/yr	PRA Initiating Events Analysis, 092-134-E-Z-0001, R0 Table 19.3.6-1
14	Inadvertent operation of a Relief Valve	2.7E-2/yr	PRA Initiating Events Analysis 092-134-E-Z-0001, R1 Table 19.1.1-1
15	Electrical buswork fails during operation	2.0E-7/hr	ALWR Volume II, Chapter 1 Annex A, page A.A-28
16	Inverter fails during operation	2.0E-5/hr	ALWR Volume II, Chapter 1 Annex A, page A.A-28
17	Initiating events caused by stuck open SRV between 1988 and 1995	4.6E-2/yr	NUREG/CR-5750. Initiating Events Study, page11, page D-40
18	Loss of Condenser Vacuum	0.2/yr	PRA Initiating Events Analysis 092-134-E-Z-0001, R0 Table 19.3.6-1
19	Spurious Actuation 15Aof a single DPV	4.9E-4/yr	PRA Initiating Events Analysis 092-134-E-Z-0001, R1 Table 2.1.2.2

Table 15A-4

Feedwater Heaters and Valves That Can Experience a Level Transmitter Failure

<table>
<tr><th></th><th>Number of strings</th><th>Number of stages</th><th>Number of heaters</th><th>Number of Extraction steam valves</th><th>Number of Extraction FWH Bypass Lines per train/total</th><th>FW Isolation valves2 train/total</th></tr>
<tr><td>LP heater stages 1-2</td><td>3</td><td>2</td><td>6</td><td>0</td><td rowspan="2">1/3</td><td rowspan="2">2/6</td></tr>
<tr><td>LP heater stages 3-4</td><td>3</td><td>2</td><td>6</td><td>6</td></tr>
<tr><td>LP direct contact feedwater heater tank, stage 5</td><td>1</td><td>1</td><td>1</td><td>3</td><td>0</td><td>0</td></tr>
<tr><td>HP heater stages 1-4</td><td>2</td><td>2</td><td>4</td><td>4</td><td>1/2</td><td>2/4</td></tr>
<tr><td>TOTAL</td><td>9</td><td>7</td><td>17</td><td>13</td><td>5</td><td>10</td></tr>
</table>

Table 15A-5

Summary of SRV Related Reactor Trip Events Between 1987 and 1993

Classification	Failure Mode	No.	LER No.
Spurious Openings	Stuck open	3	
			237/90-006
			265/91-02
			352/95-008
Testing Induced Openings			
	Closed Promptly after trip	2	397/92-033
	Closed after time delay	2	265/93-006
			373/93-002
	Stuck Open	3	254/89-004
			324/90-004
			354/87-047

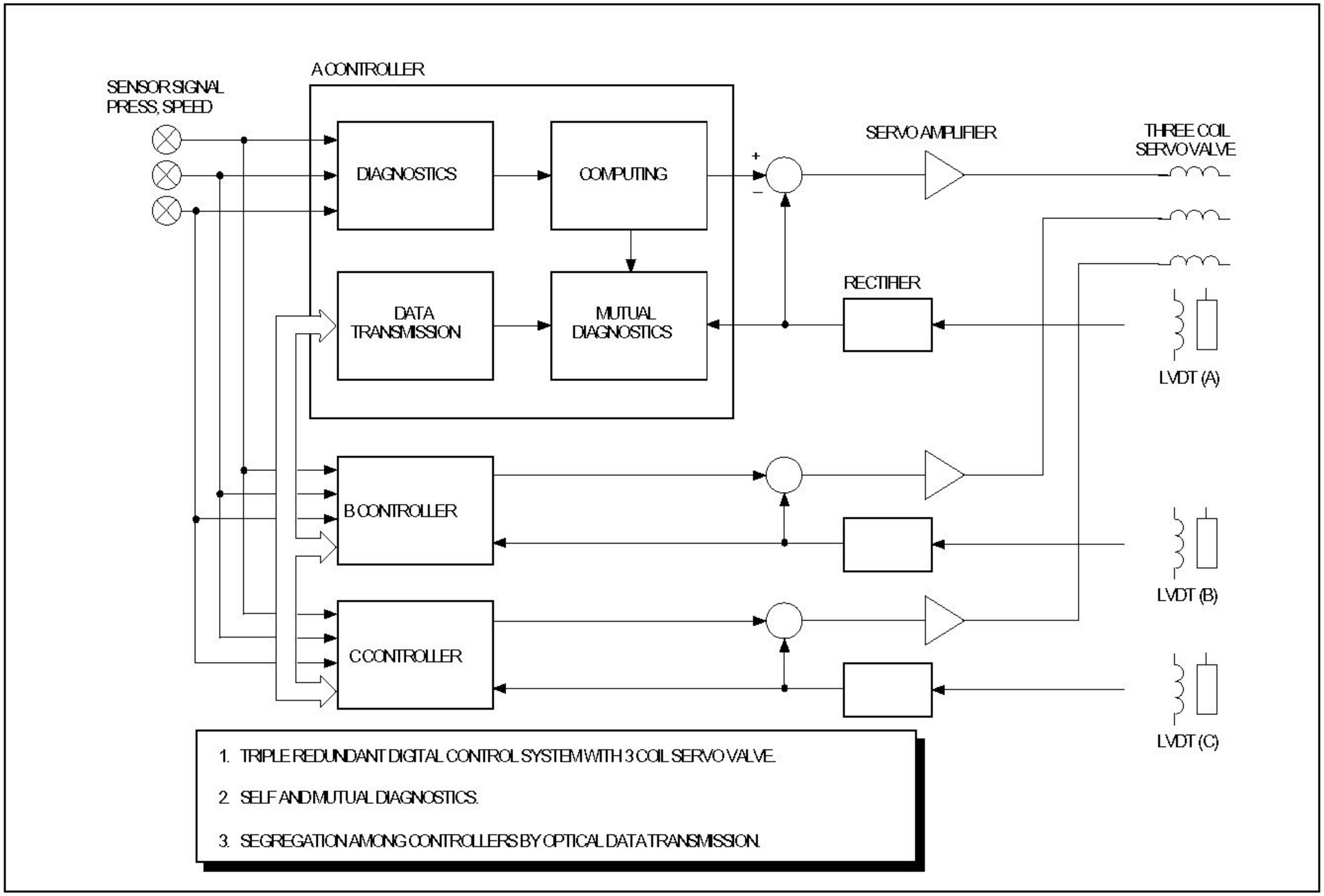


Figure 15A.1-1. Triple Redundant Control System

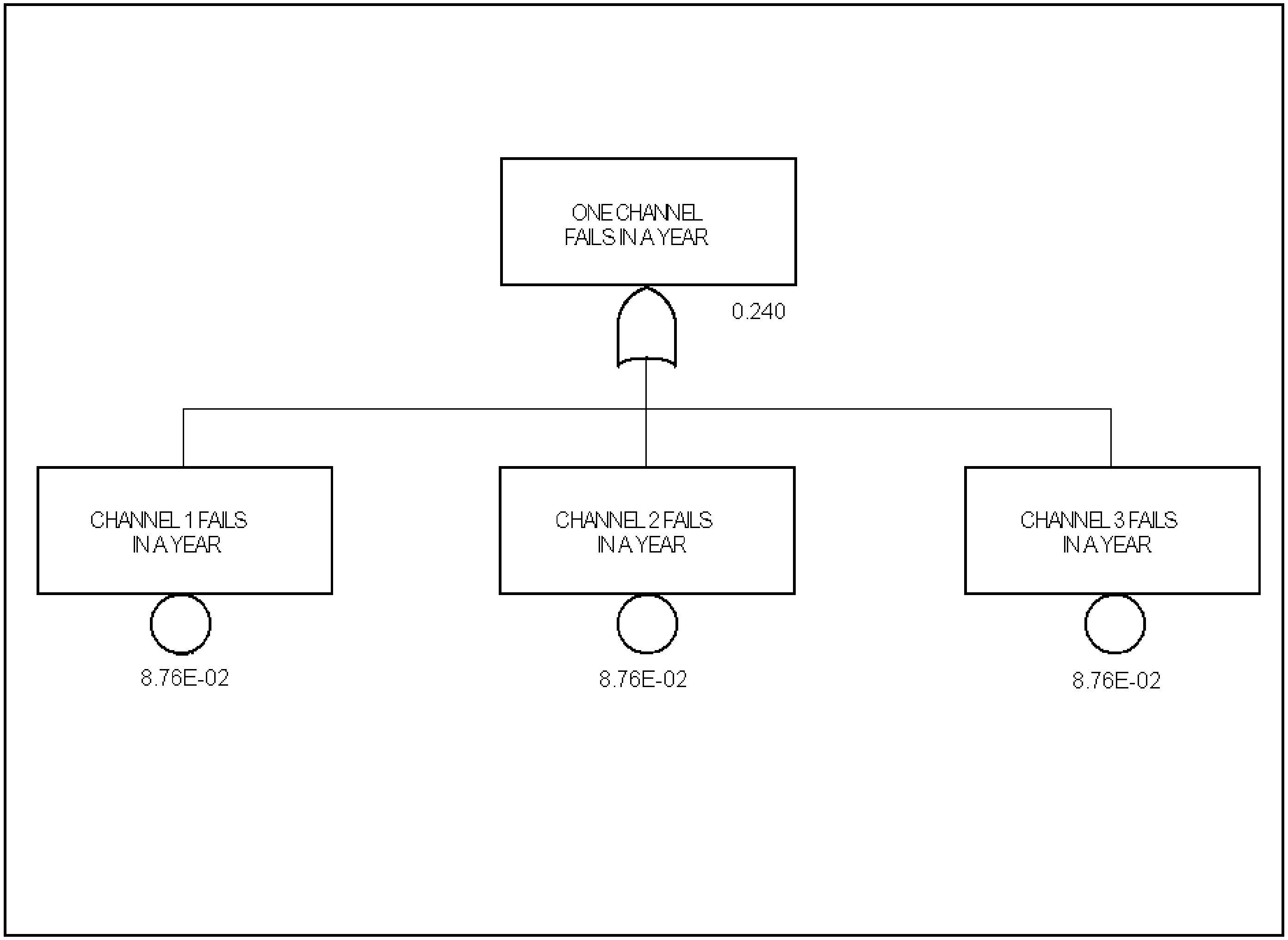


Figure 15A.1-2. Fault Tree Model of First Instrument Channel Failing – Three Channel Logic

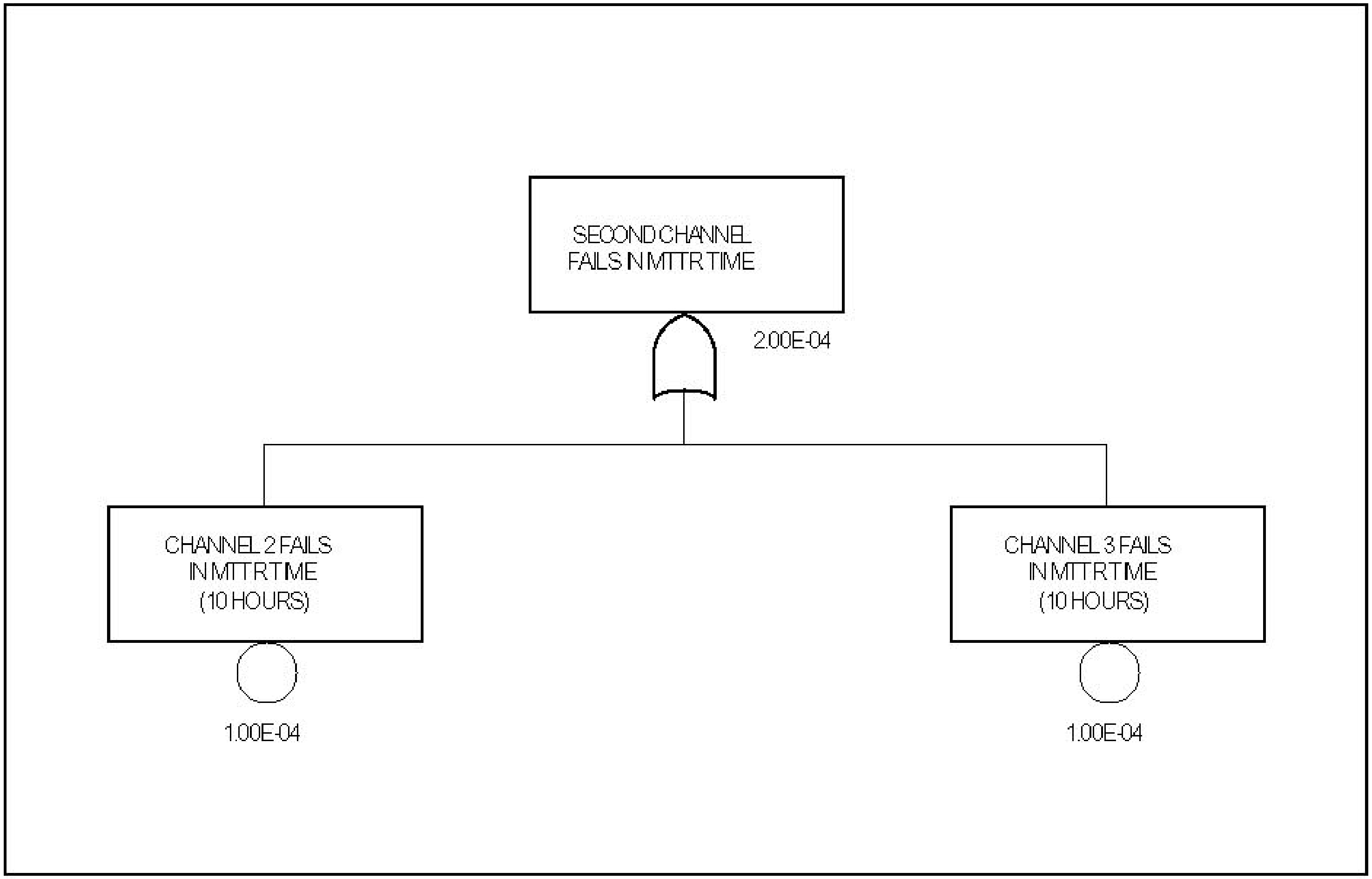


Figure 15A.1-3. Fault Tree Model of Second Instrument Channel Failing When First Failure Is Detected – Three Channel Logic

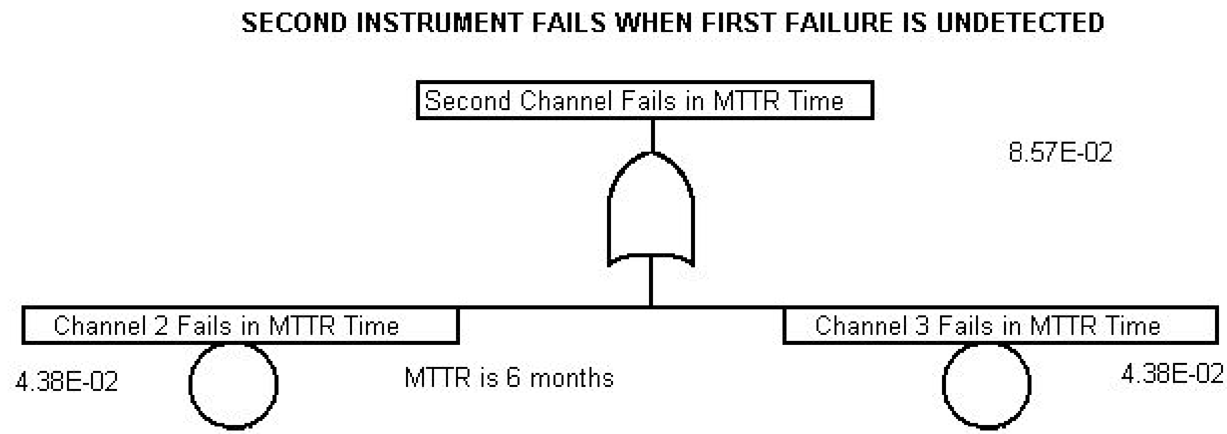


Figure 15A.1-4. Fault Tree Model of Second Instrument Channel Failing When First Failure Is Undetected – Three Channel Logic

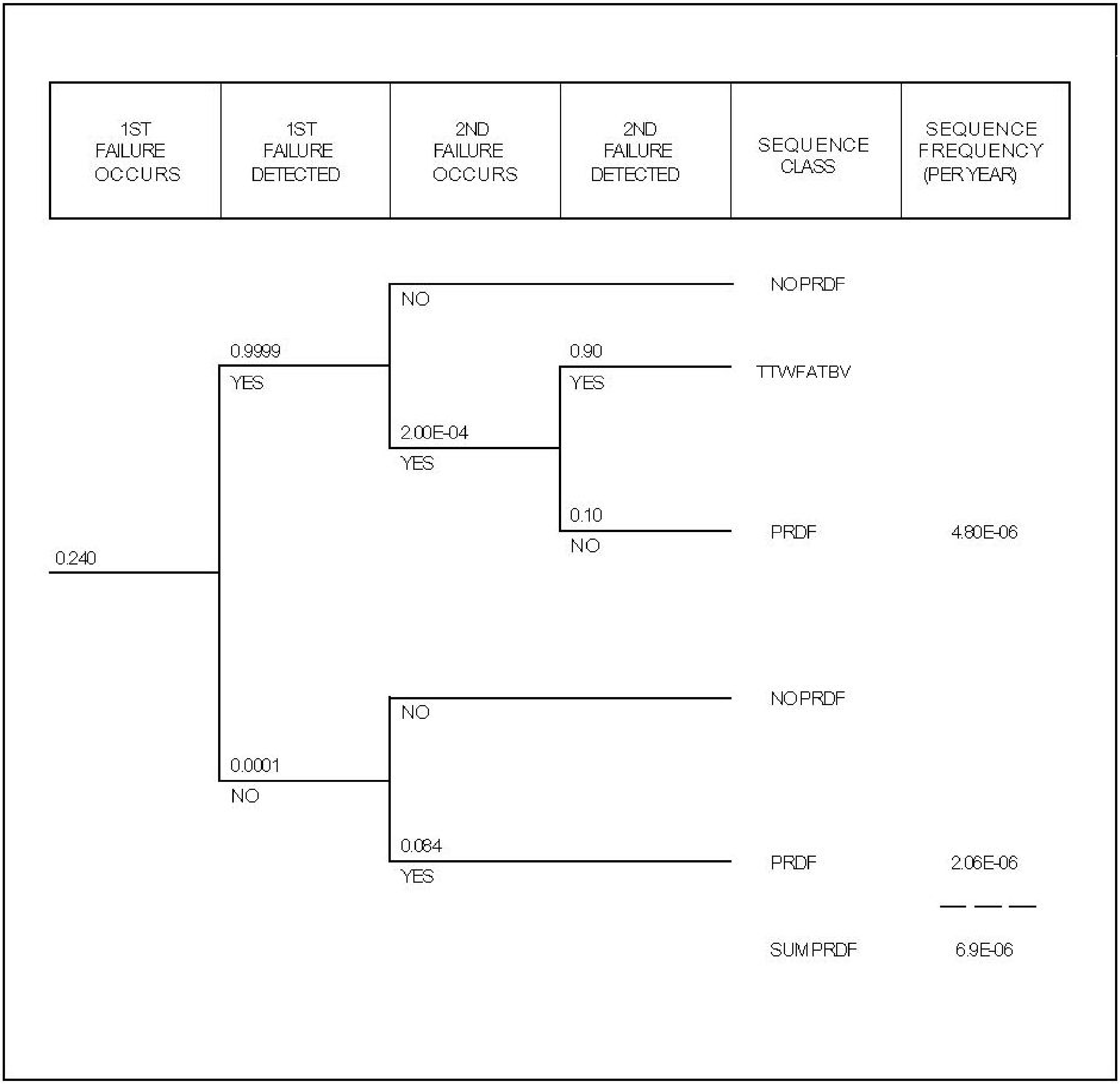


Figure 15A.1-5. Event Tree for Triplicated Digital EHC (PRDF)

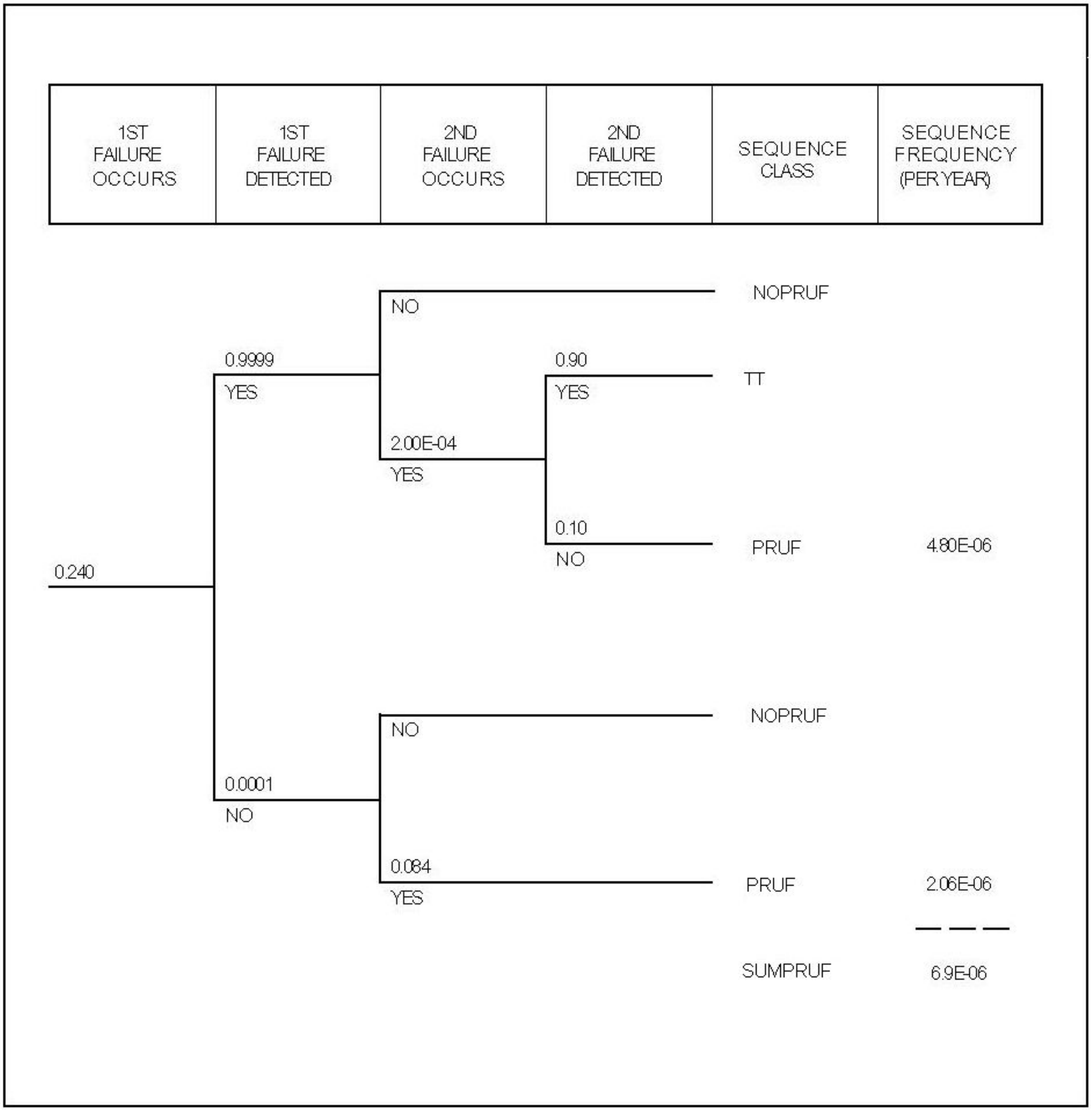


Figure 15A.1-6. Event Tree for Triplicated Digital EHC Pressure Regulator Upscale Failure (PRUF)

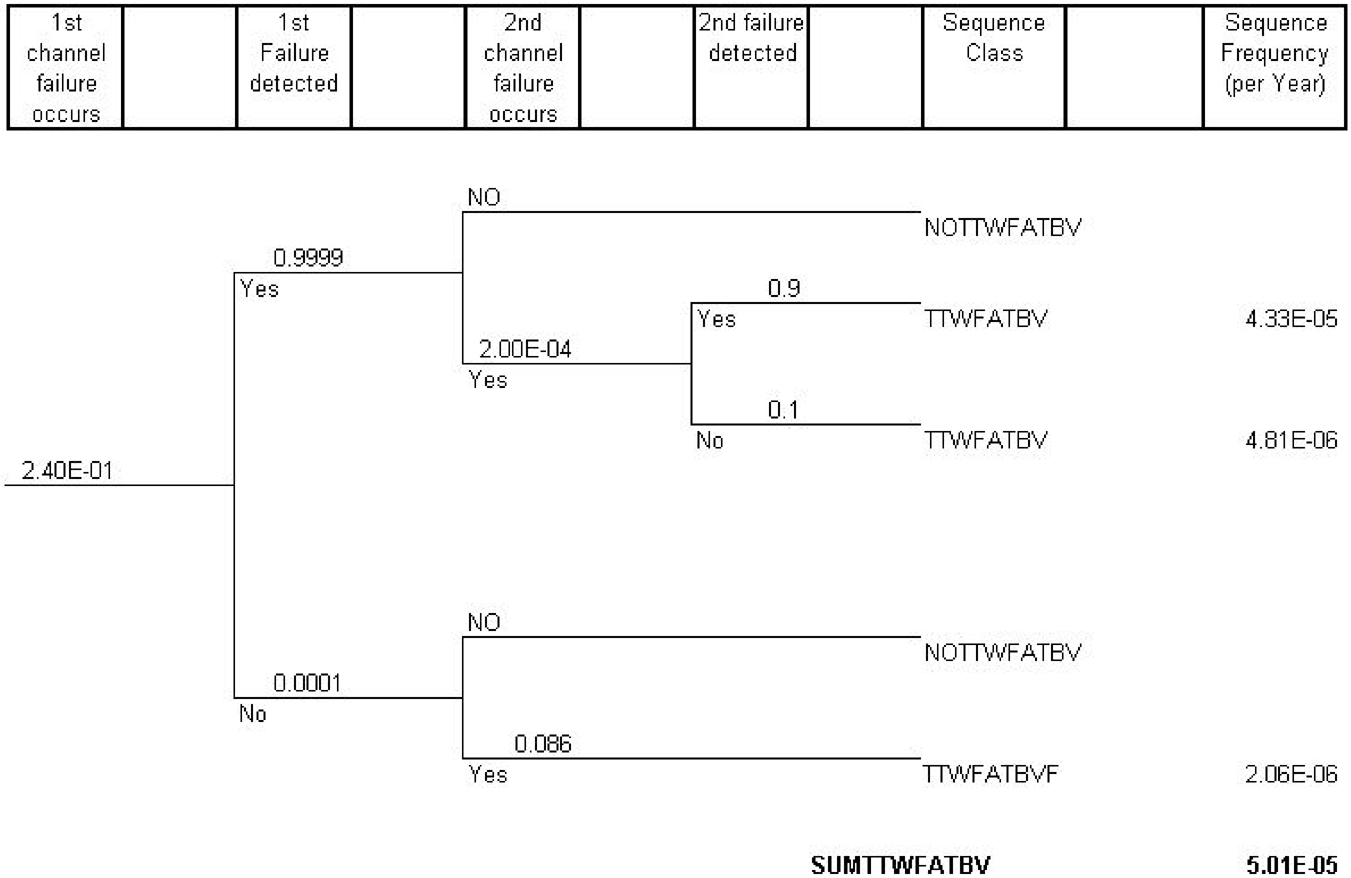


Figure 15A.2-1. Event Tree for Triplicated Digital EHC Channel Failures Turbine Trip with 100% Turbine Bypass Failure

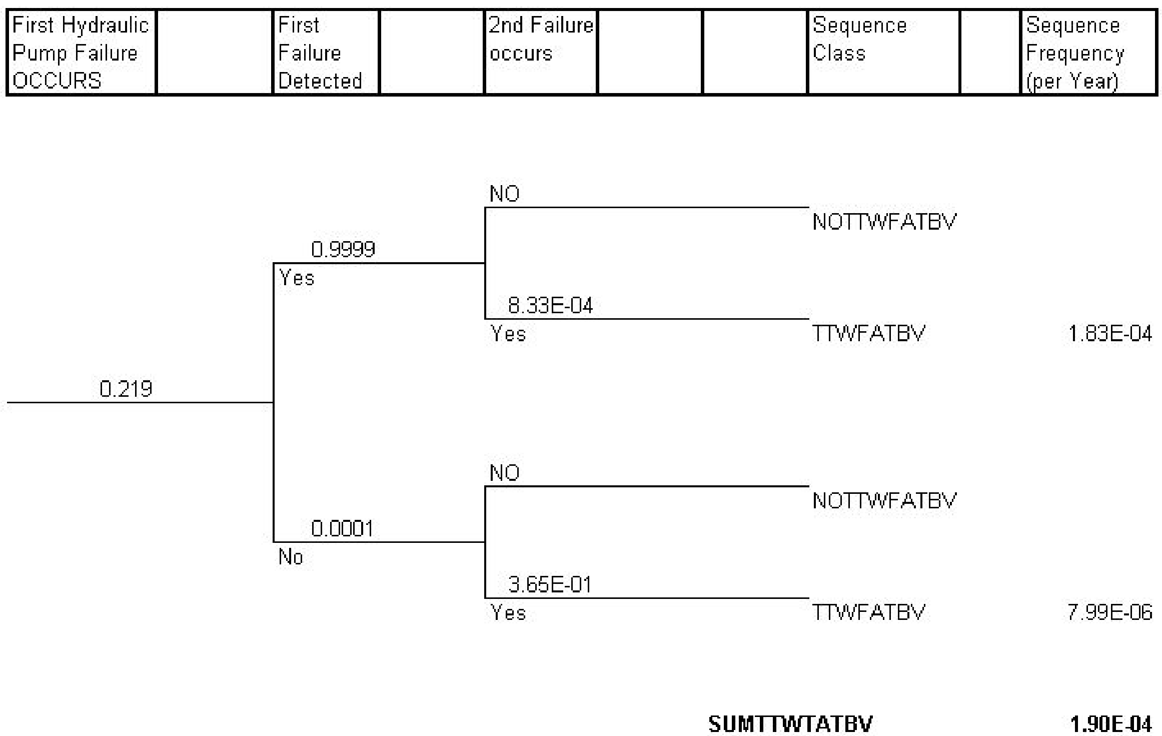


Figure 15A.2-2. Event Tree for Hydraulic Power Unit Pump Failure: Turbine Trip with 100% Turbine Bypass Failure

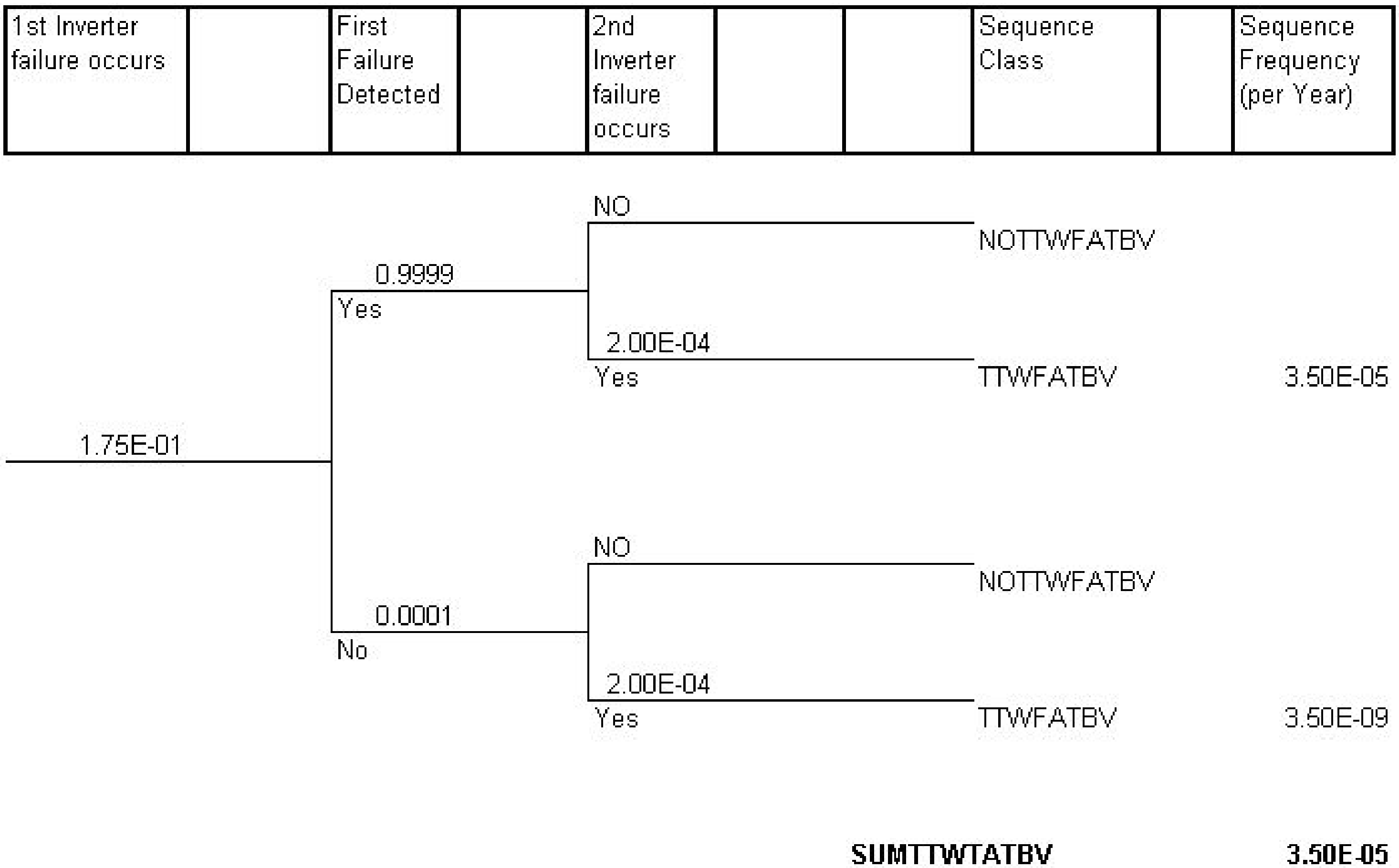


Figure 15A.2-3. Event Tree for Hydraulic Power Unit Power Failure: Turbine Trip with 100% Turbine Bypass Failure

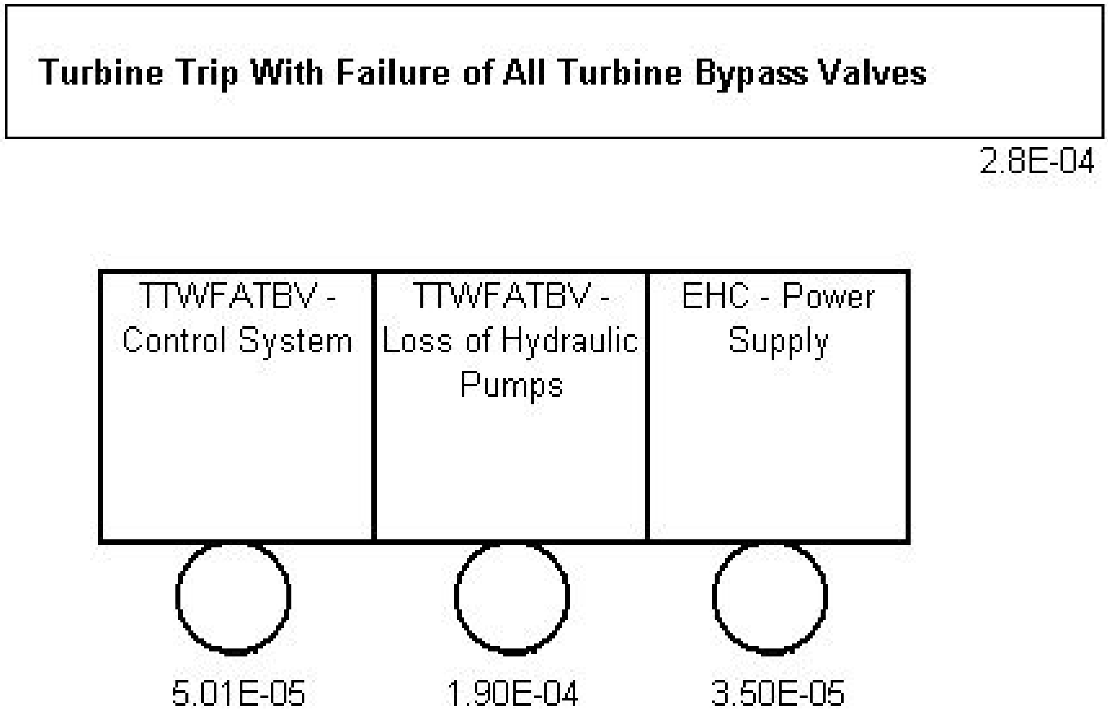


Figure 15A.2-4. Fault Tree for Turbine Trip with 100% Turbine Bypass Failure – Overall Failure Rate

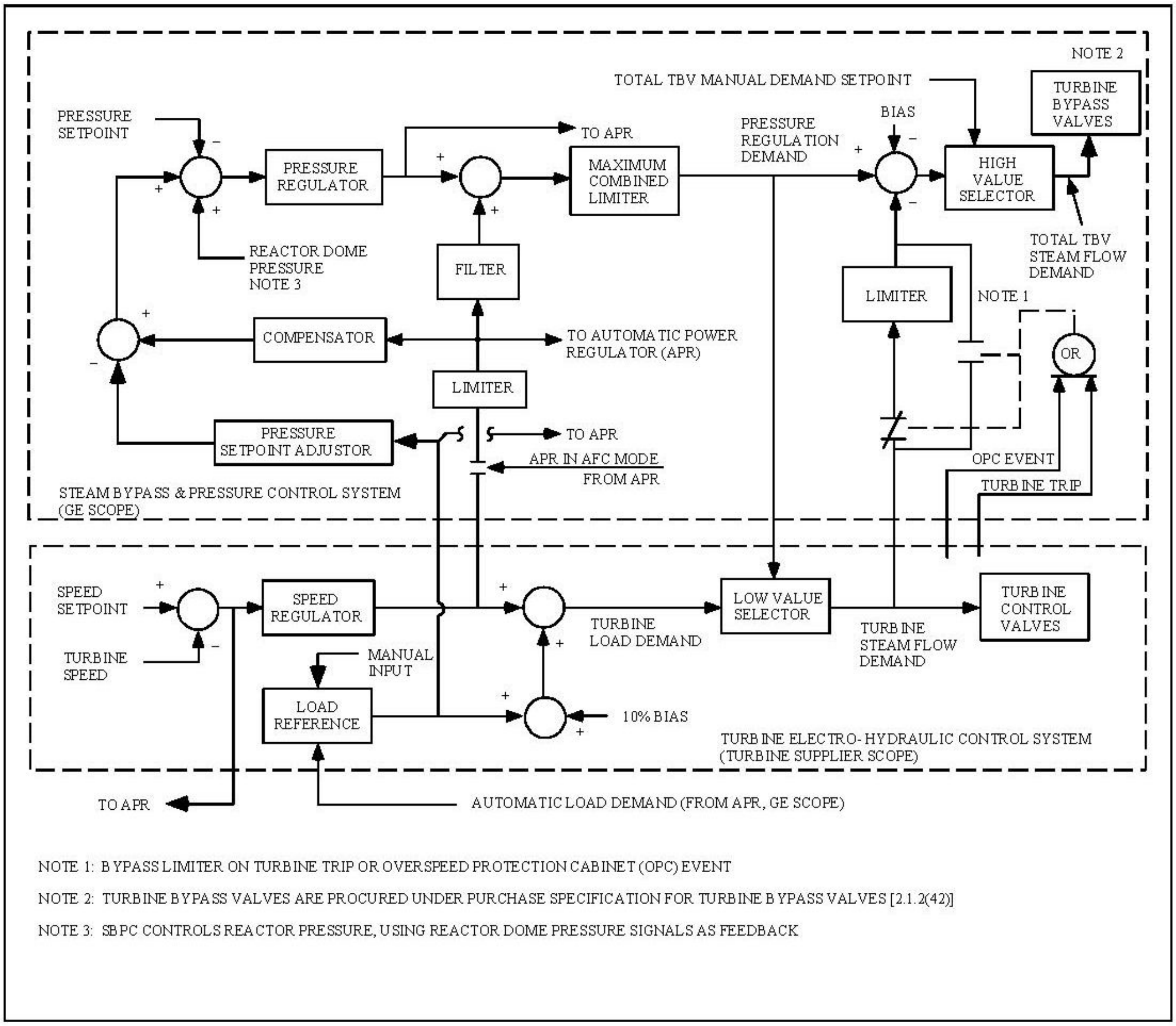


Figure 15A.2-5. SB&PC Functional Block Diagram

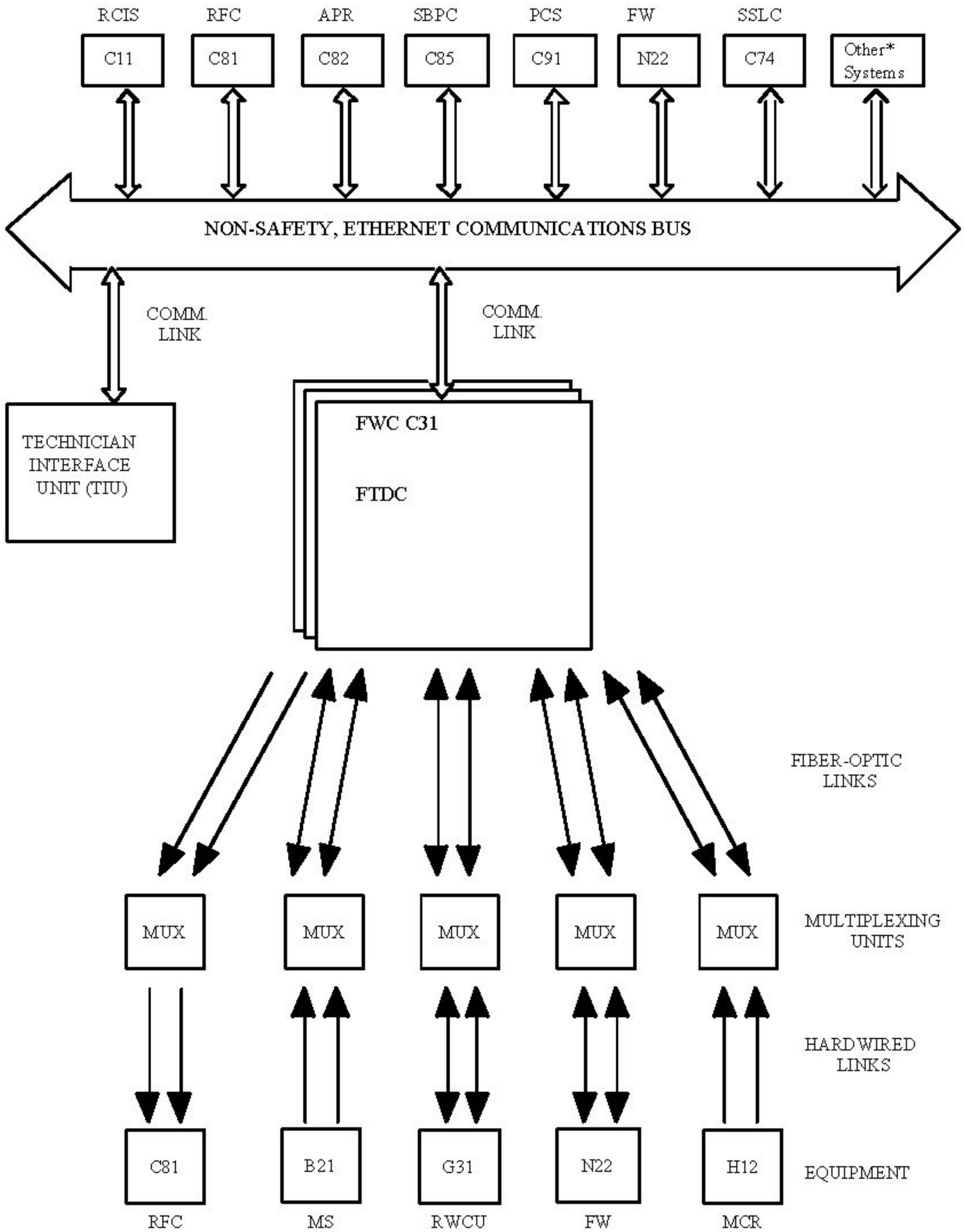


Figure 15A.3-1. Feedwater Control System Configuration

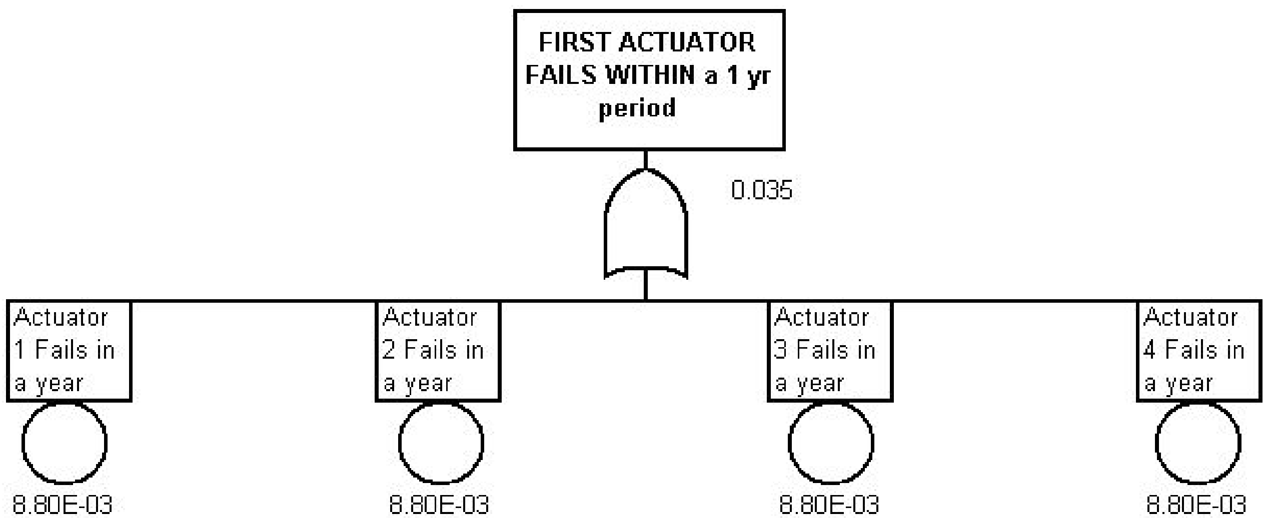


Figure 15A.3-2. Fault Tree Model of First Feedwater Pump Actuator Failing

1st failure occurs (Actuator)		1st Failure detected		2nd Failure occurs (electronic module)		2nd failure detected		Sequence Class		Sequence Frequency (per year)

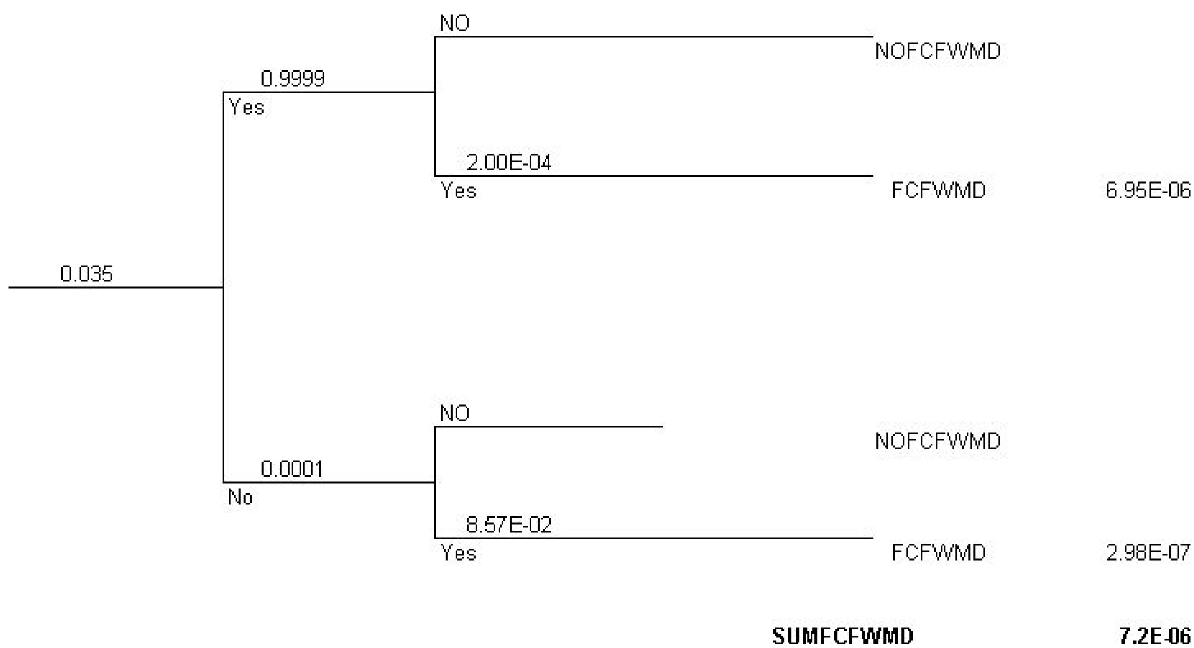


Figure 15A.3-3. Event Tree for Triplicated Digital Control: Feedwater Control Failure – Maximum Demand

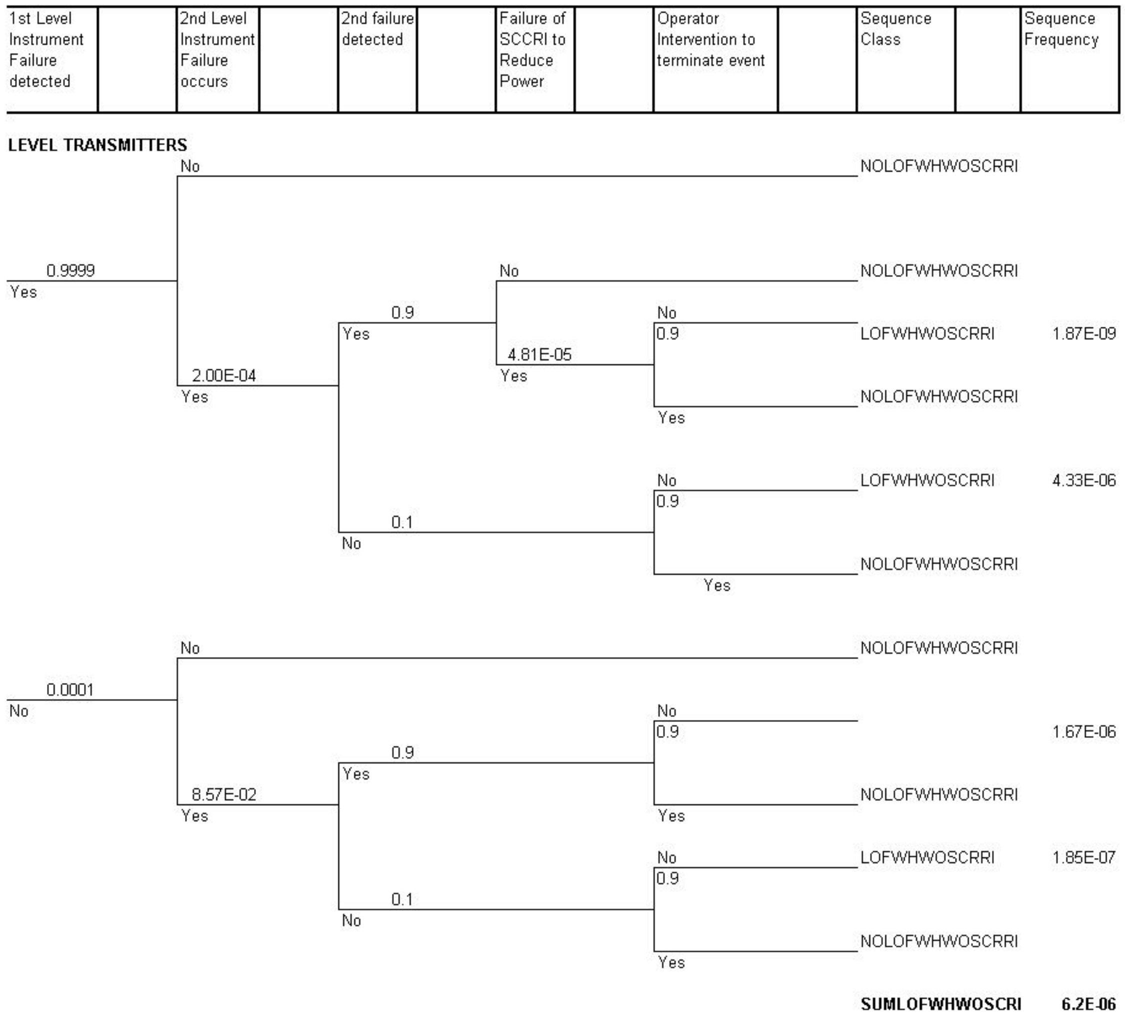


Figure 15A.4-1. Event Tree for Triplicated Digital Control Loss of Feedwater Heating With Failure of Selected Control Rod Run-In

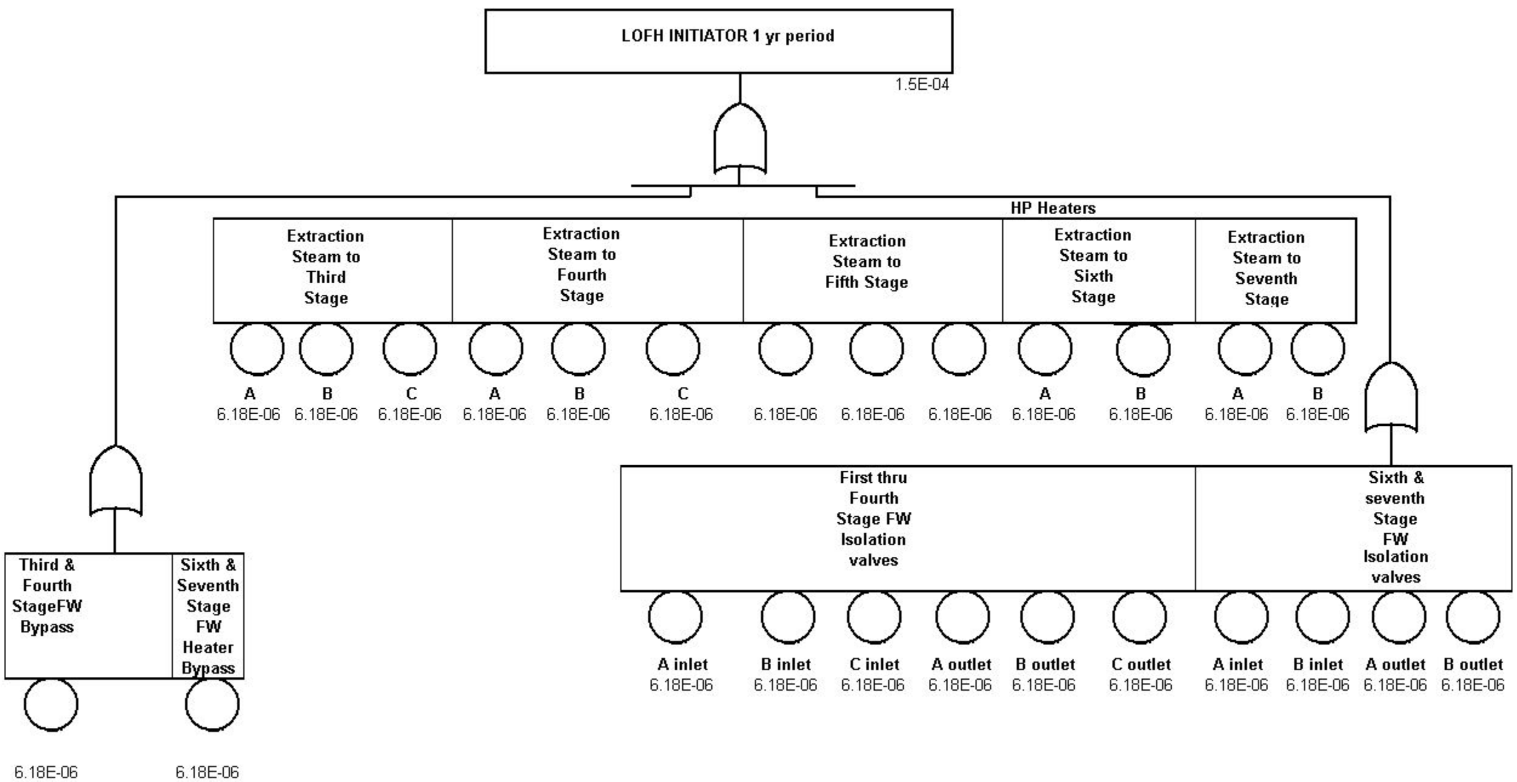


Figure 15A.4-2. Fault Tree for Triplicated Digital EHC Loss of Feedwater Heating With Failure of Selected Control Rod Run-In From Any of the FWH

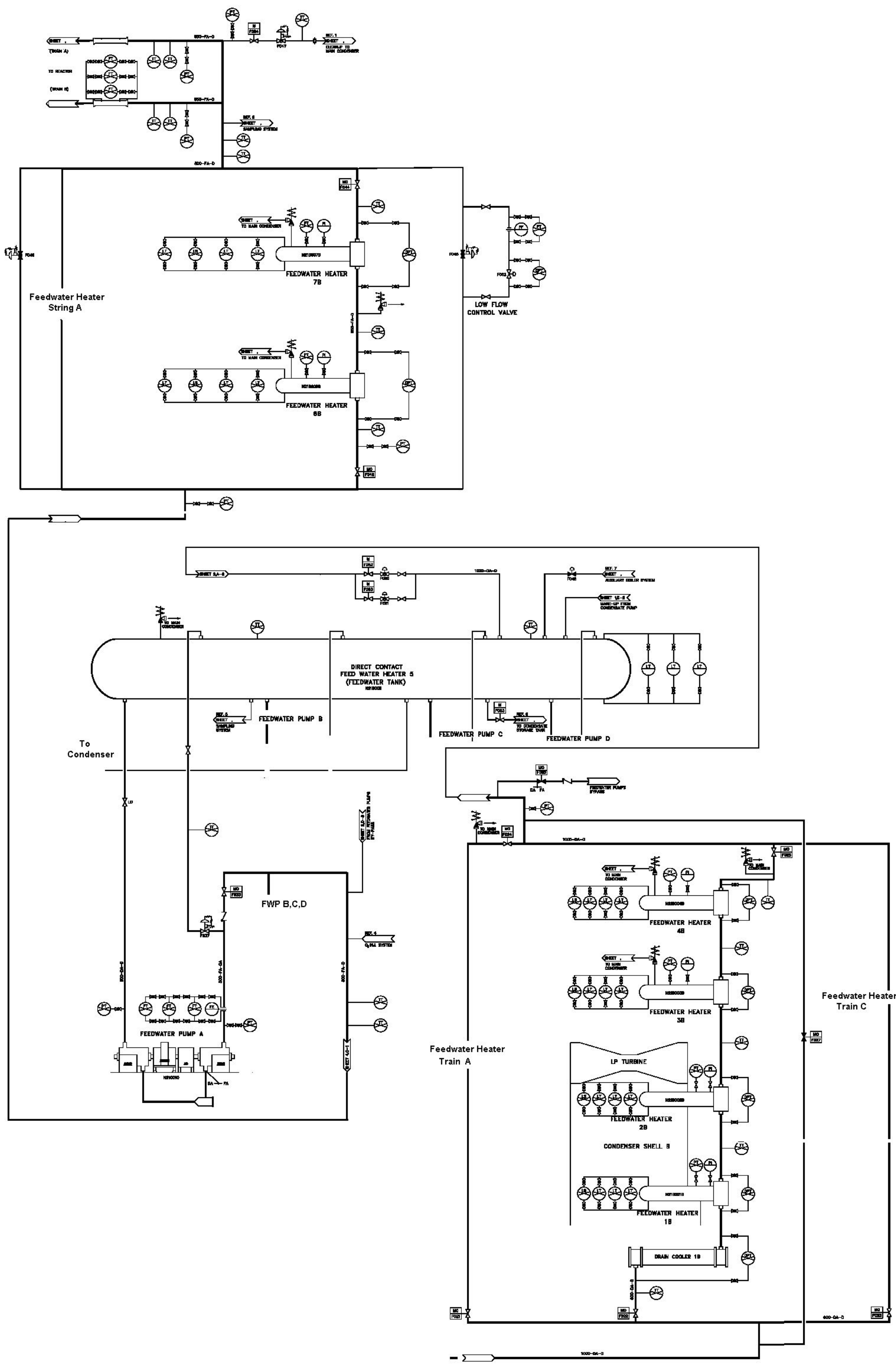


Figure 15A.4-3. Simplified Flow Diagram of Feedwater Heaters

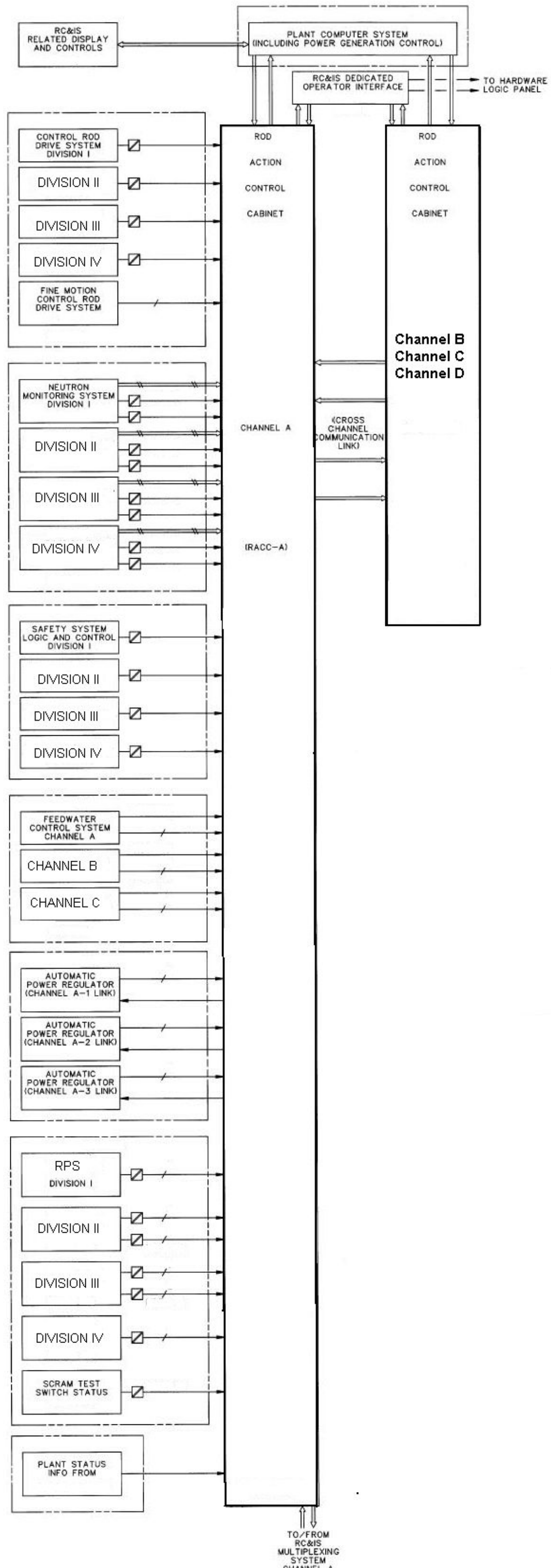


Figure 15A.4-4. Simplified Diagram of the Digital RCIS

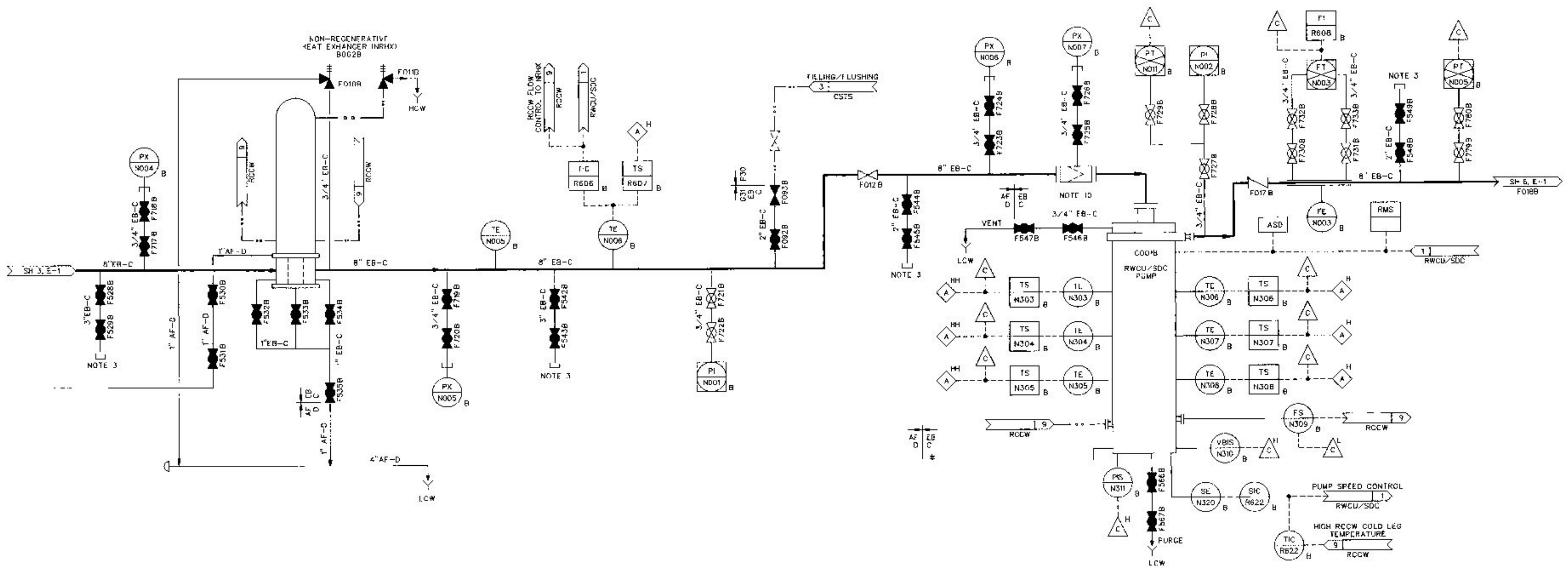


Figure 15A.5-1. RWCU/SDC Simplified P&ID

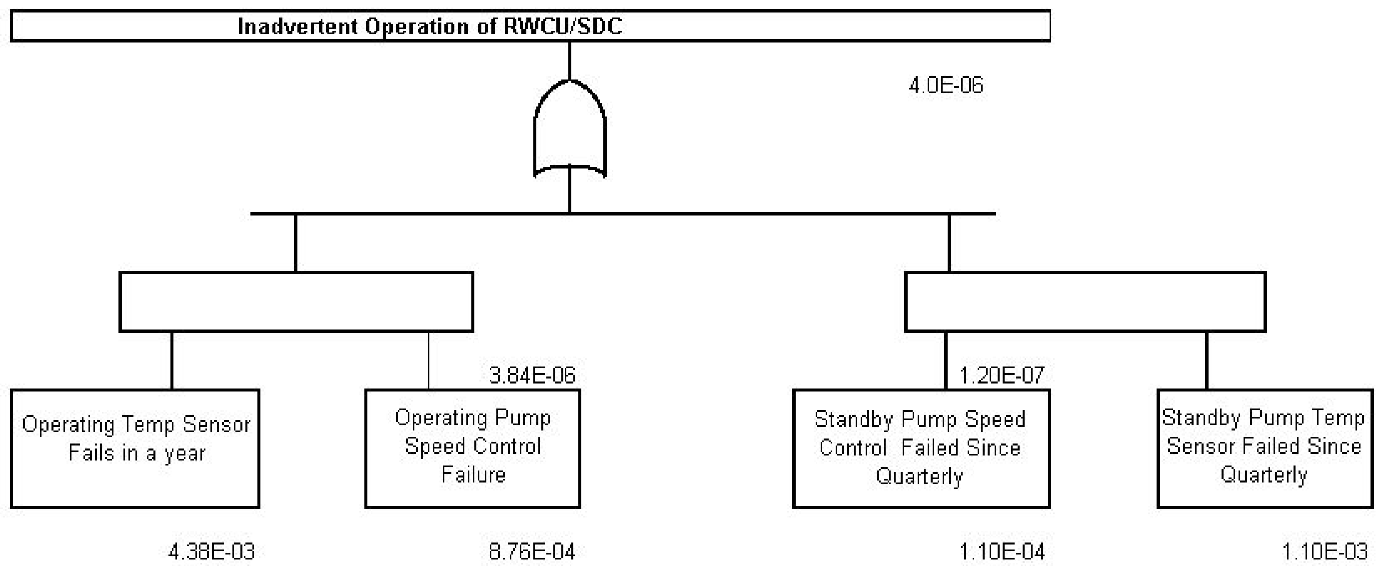


Figure 15A.5-2. Fault Tree for Inadvertent Operation of RWCU/SDC

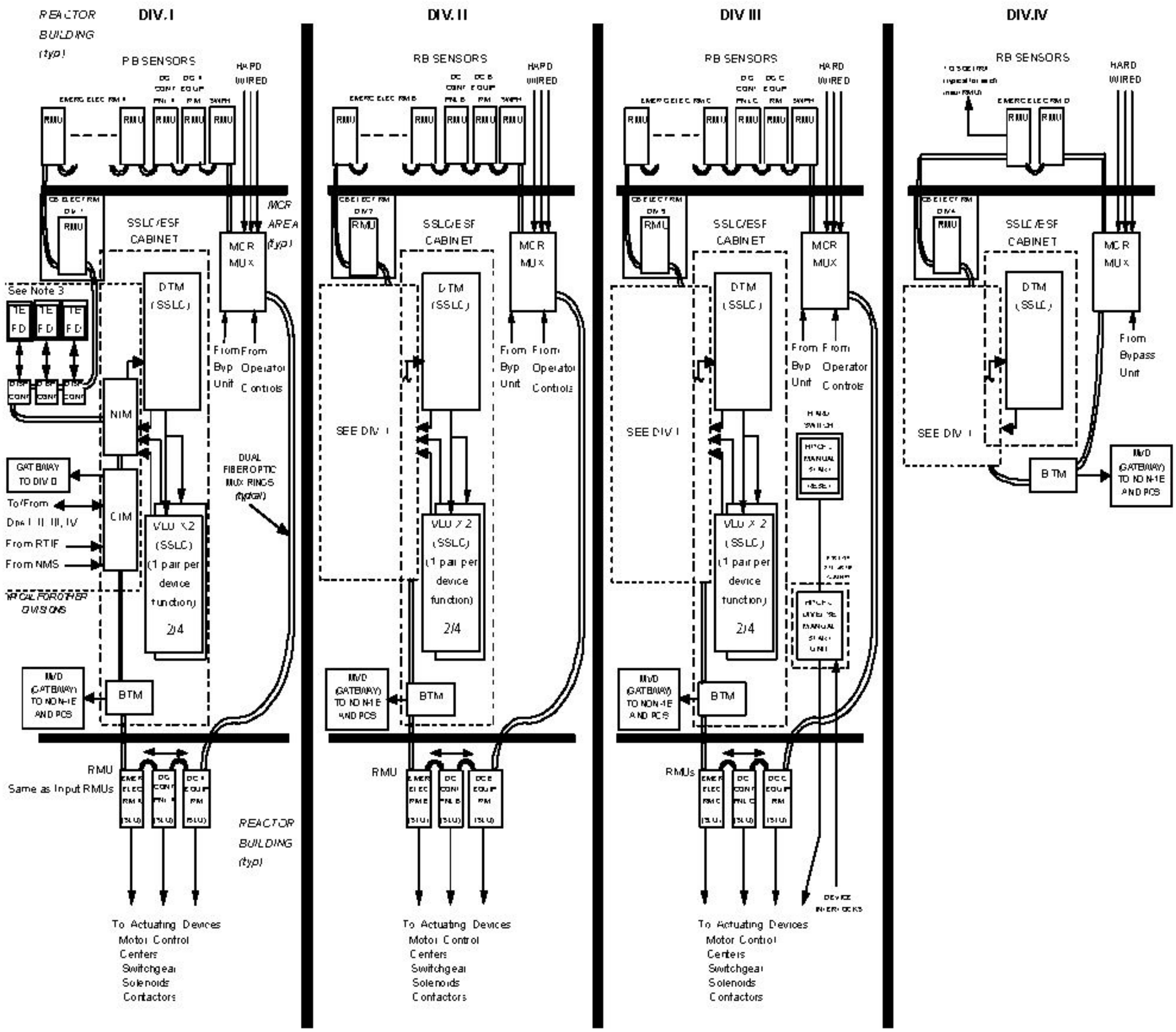


Figure 15A.6-1. SSLC Actuation Logic

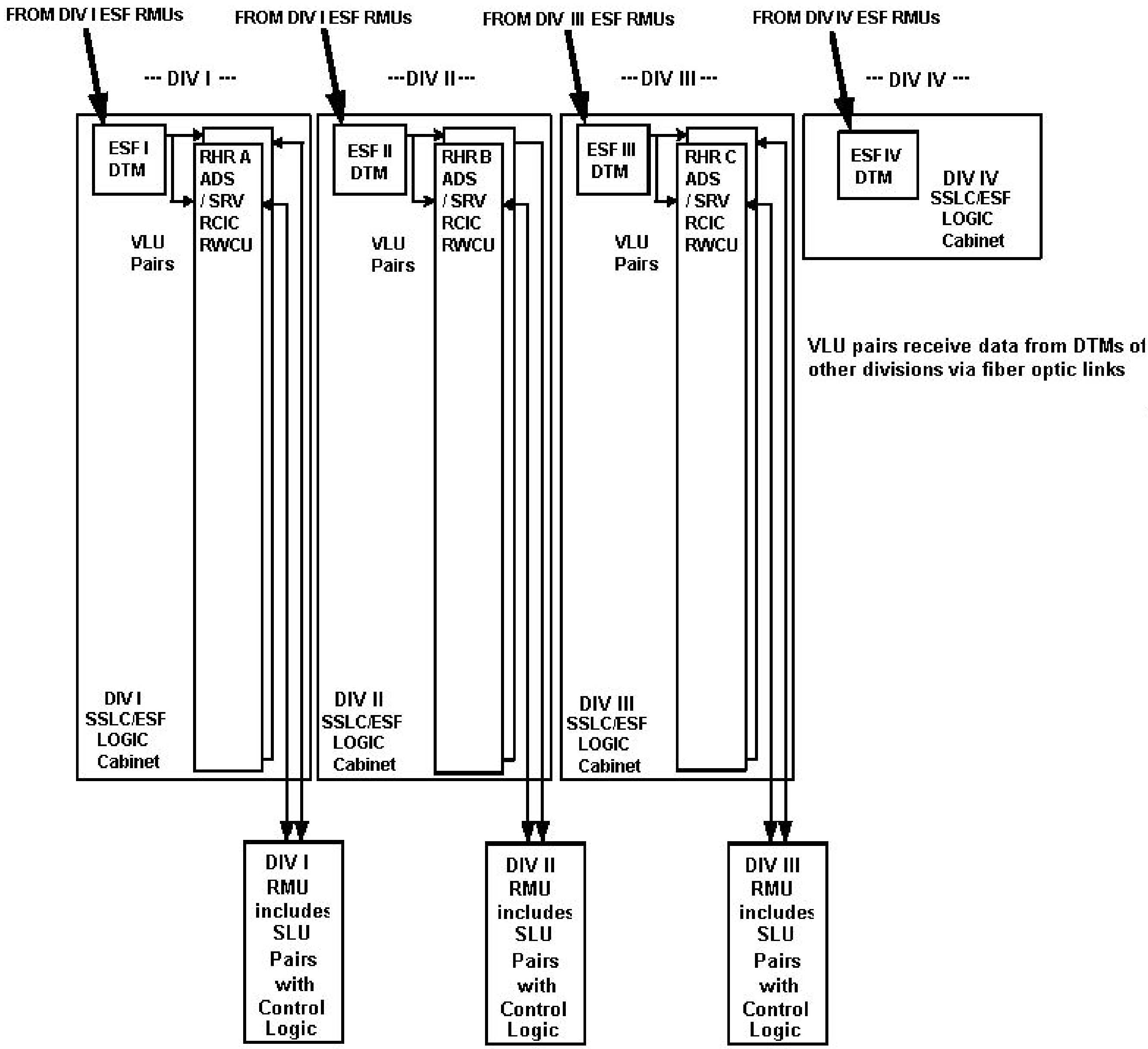


Figure 15A.6-2. SSLC RMU Actuation Logic

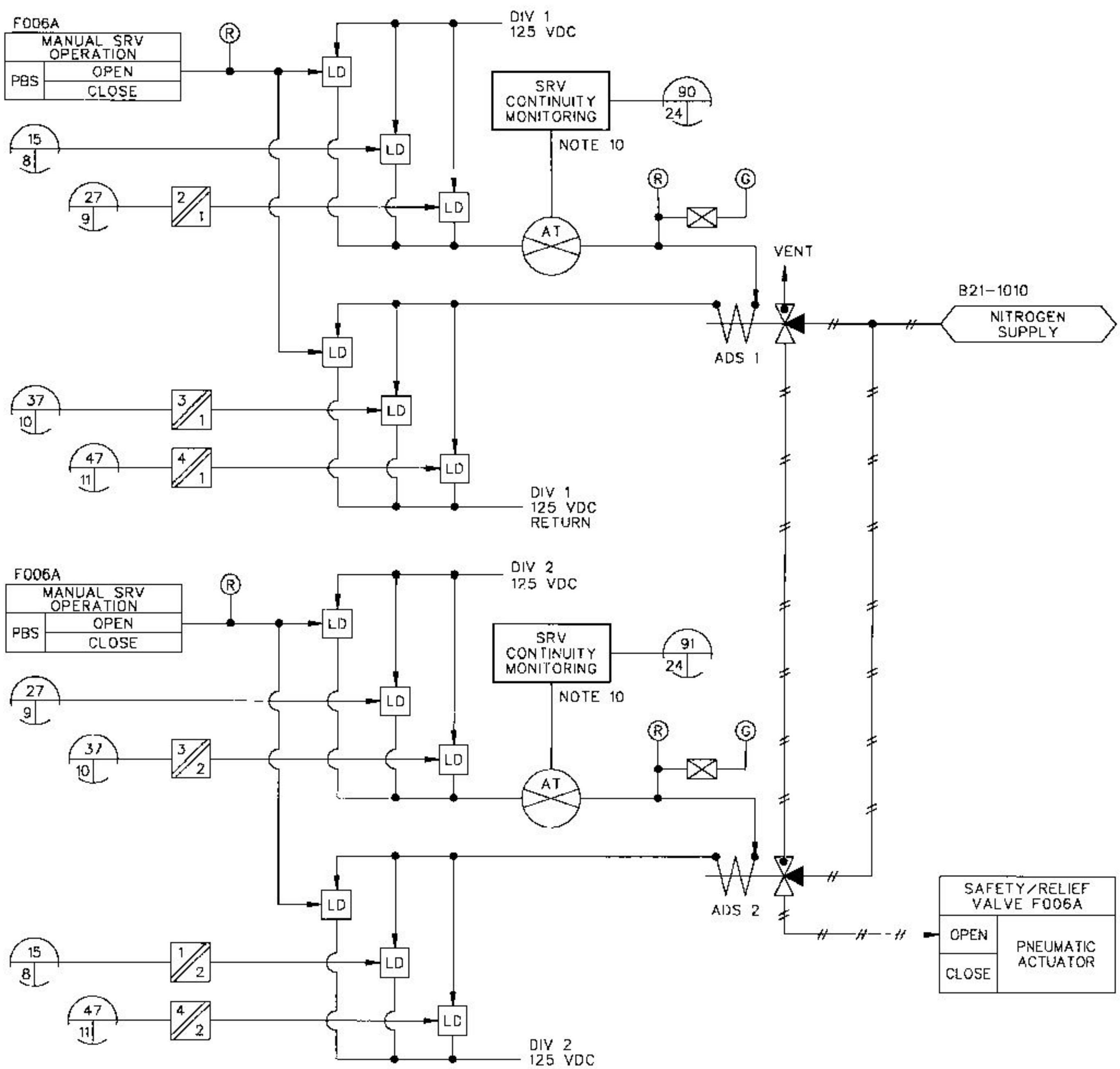


Figure 15A.6-3. SRV Actuation Logic

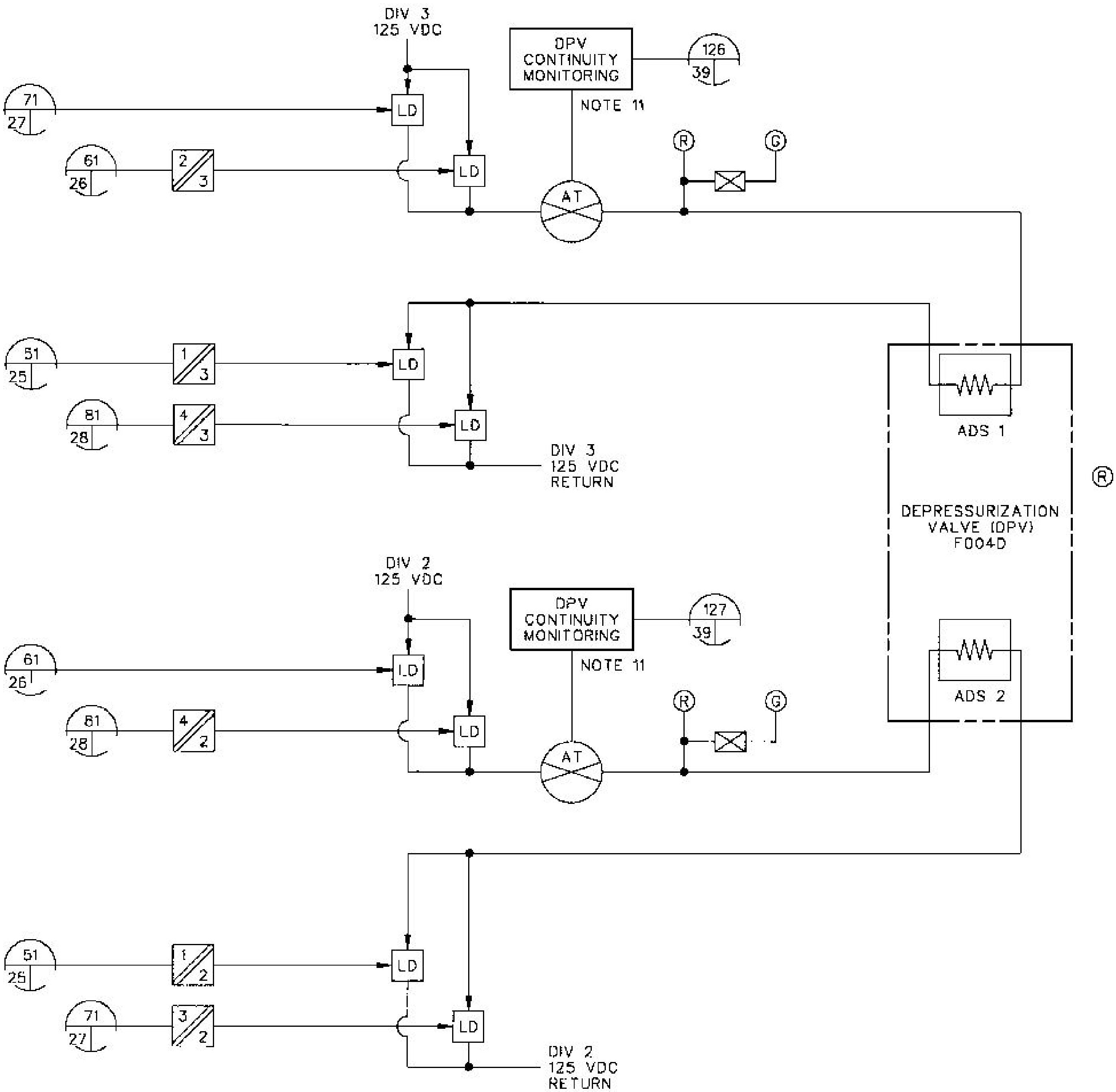


Figure 15A.6-4. DPV Actuation Logic

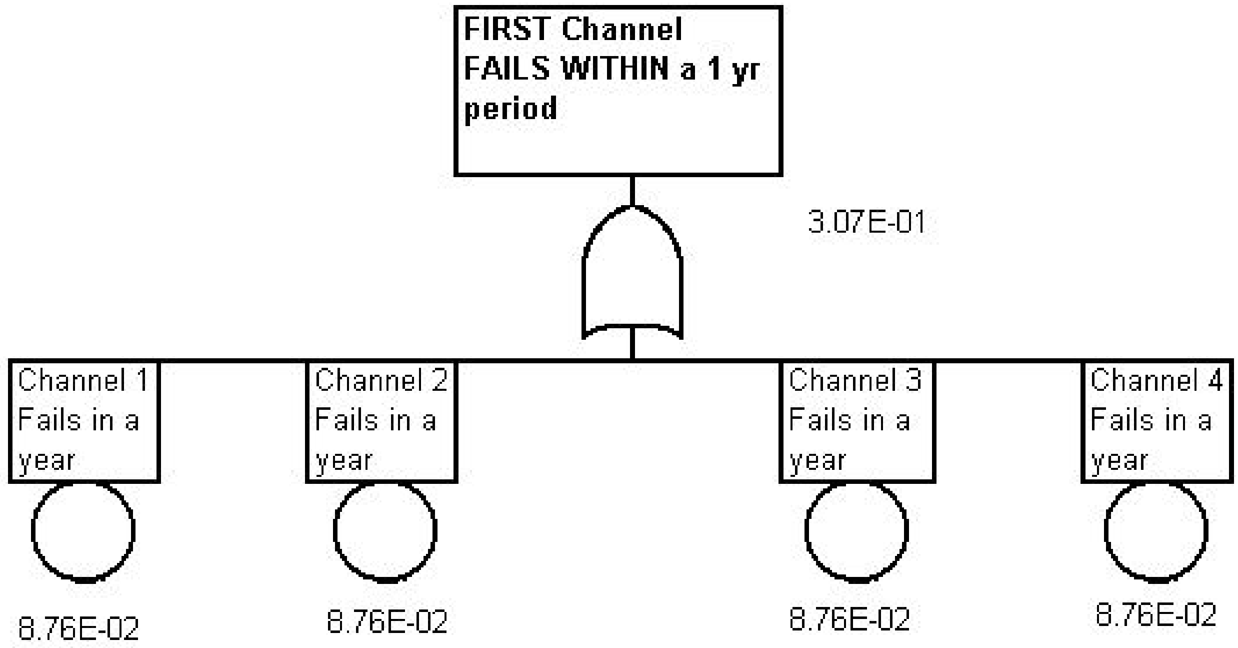


Figure 15A.6-5. Fault Tree Model of First Instrument Channel Failing – Four Channel Logic

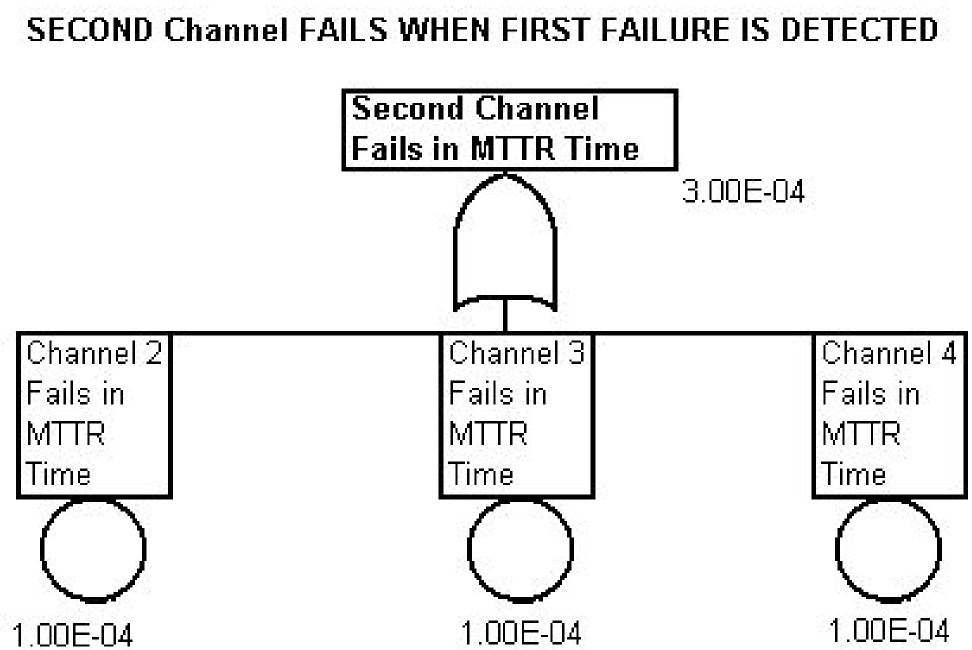


Figure 15A.6-6. Fault Tree Model of Second Instrument Channel Failing When First Failure Is Detected – Four Channel Logic

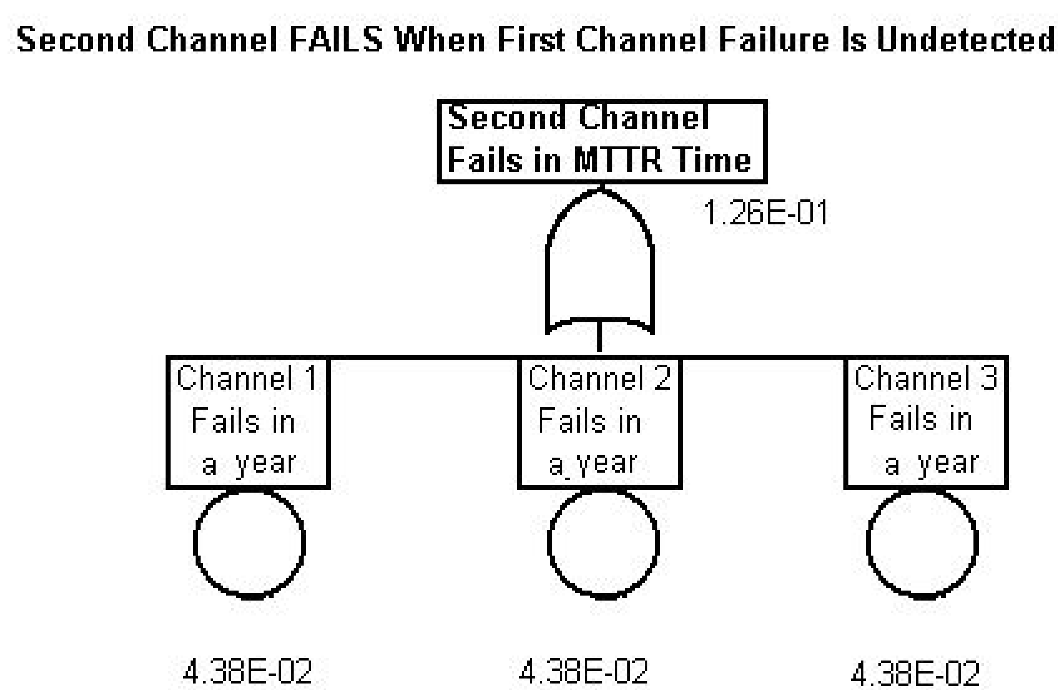


Figure 15A.6-7. Fault Tree Model of Second Instrument Channel Failing When First Failure Is Undetected – Four Channel Logic

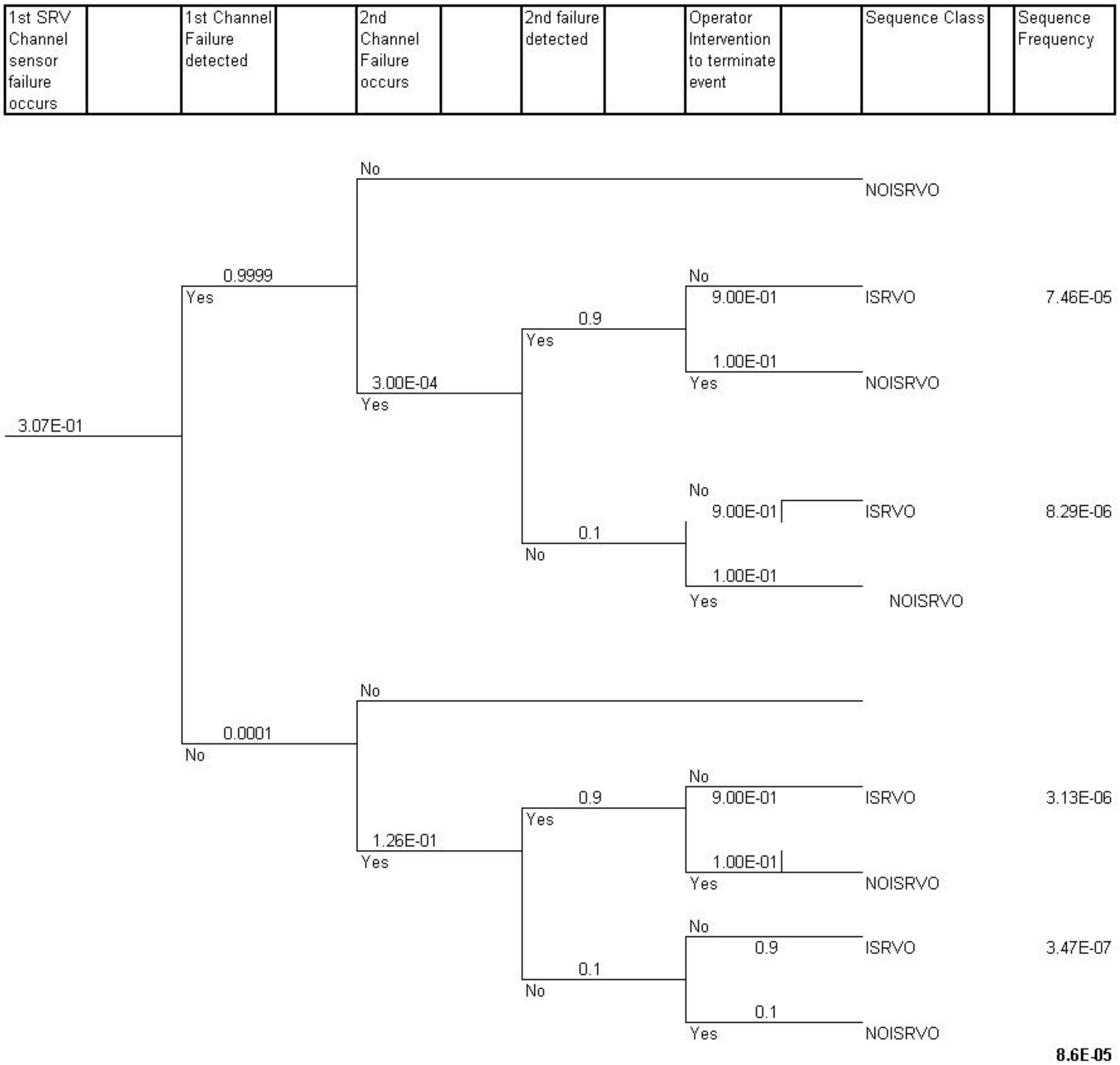


Figure 15A.6-8. Event Tree for Dual Redundant Digital ISRVO – Logic Initiated Openings

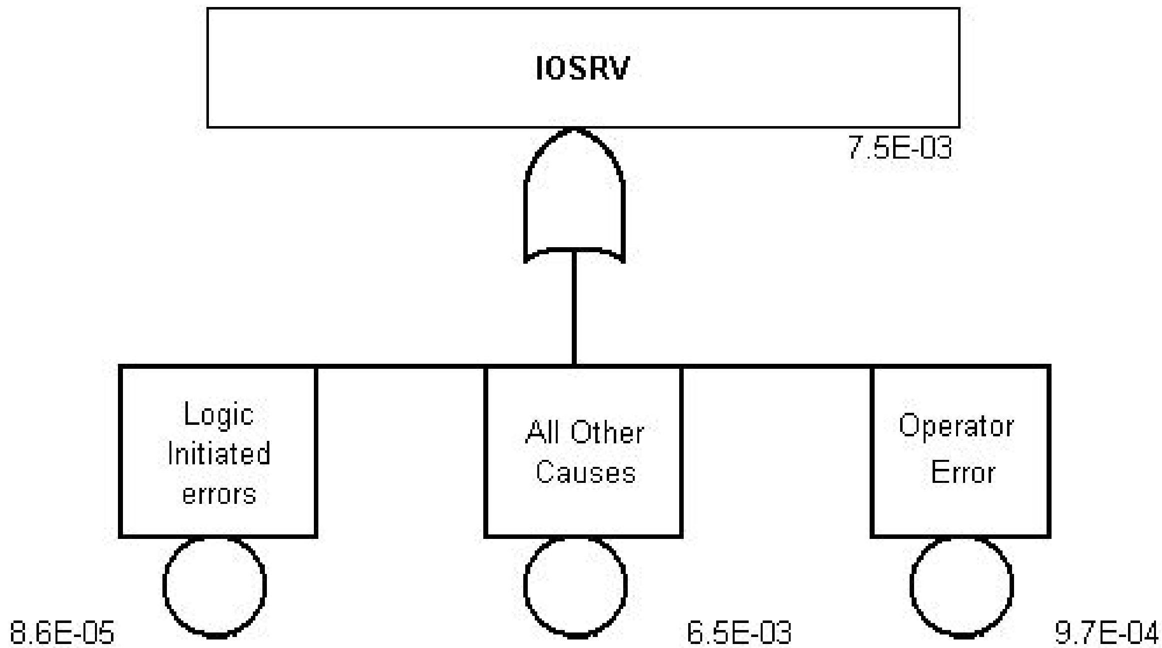


Figure 15A.6-9. Fault Tree for Inadvertent Opening of an SRV (IOSRV) – Overall Failure Probability

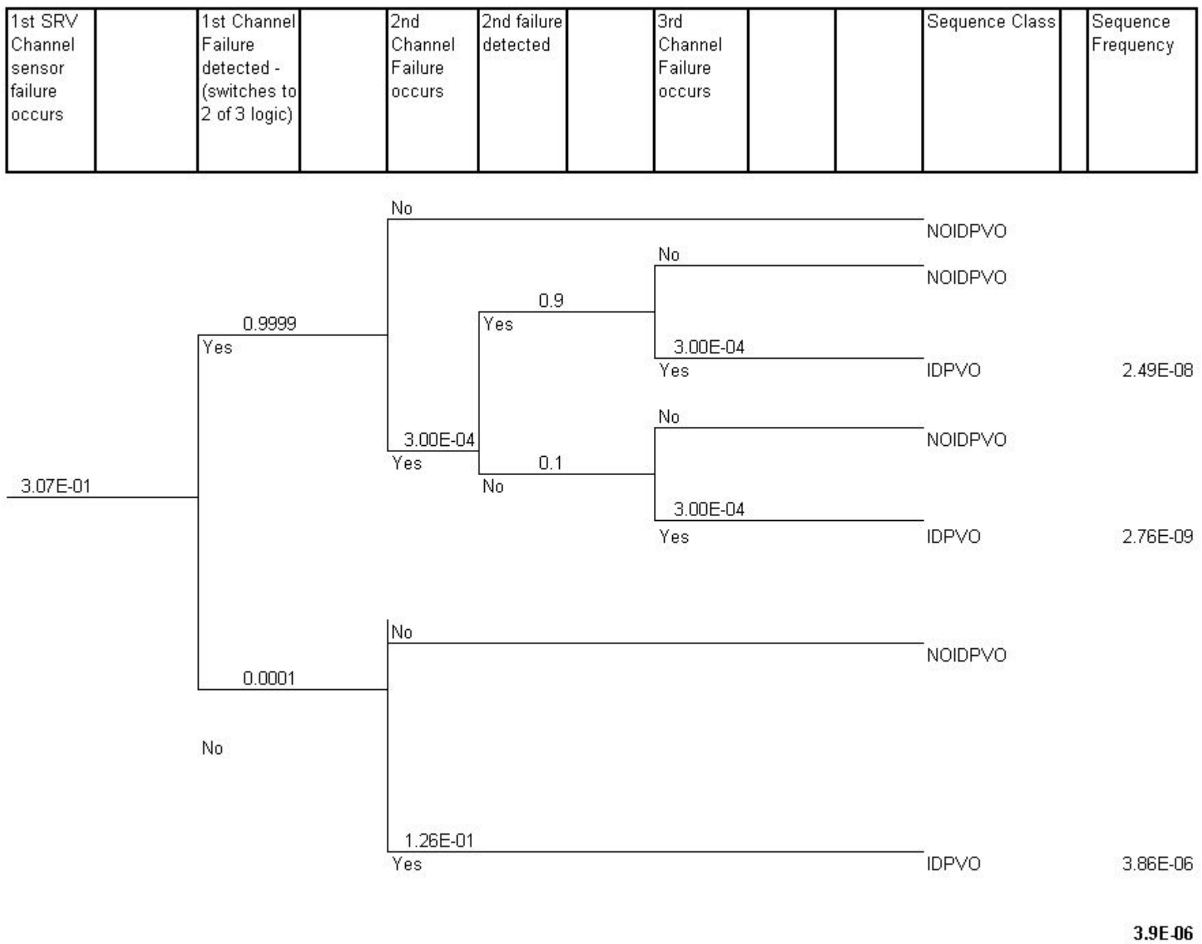


Figure 15A.6-10. Fault Tree for Dual Redundant Digital IDPVO – Logic Initiated Openings

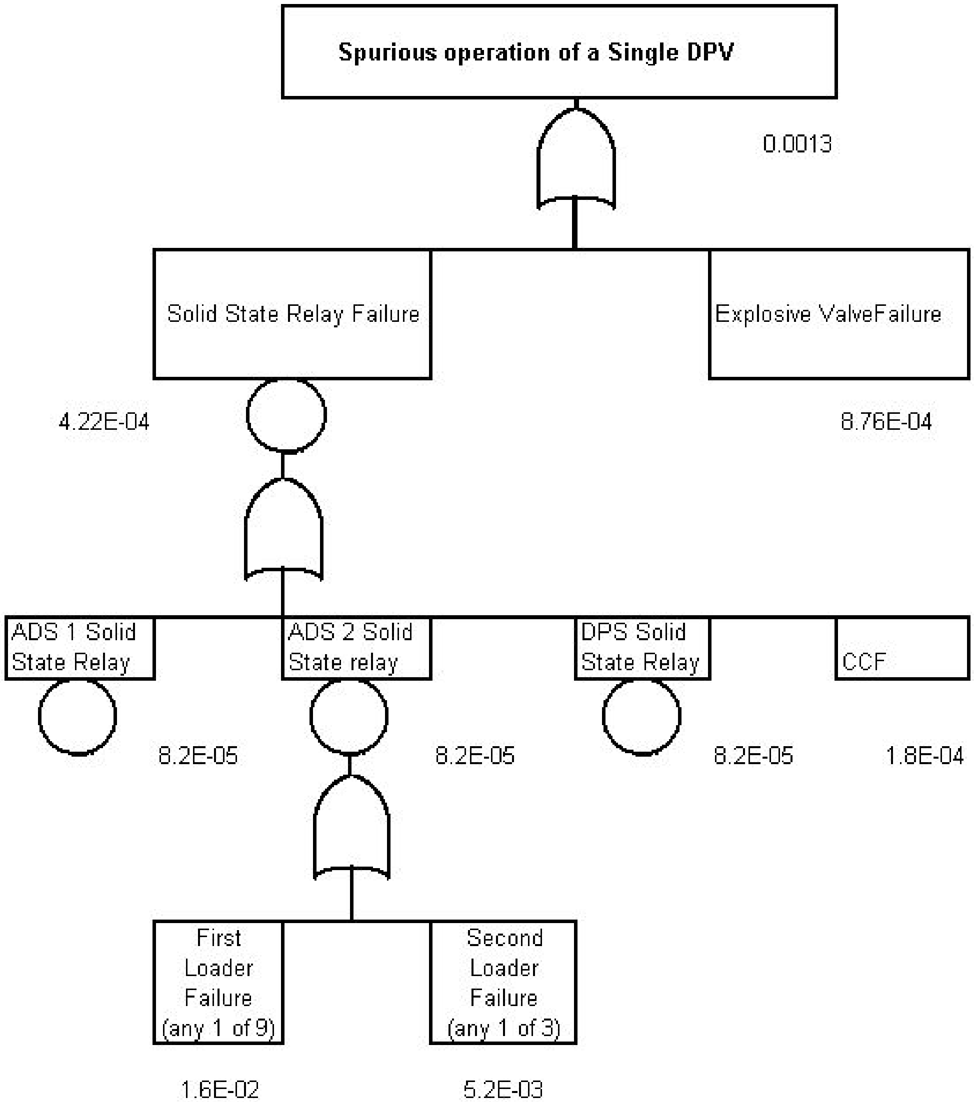


Figure 15A.6-11. Fault Tree for Dual Redundant Digital IDPVO – Single Valve Failure Probability

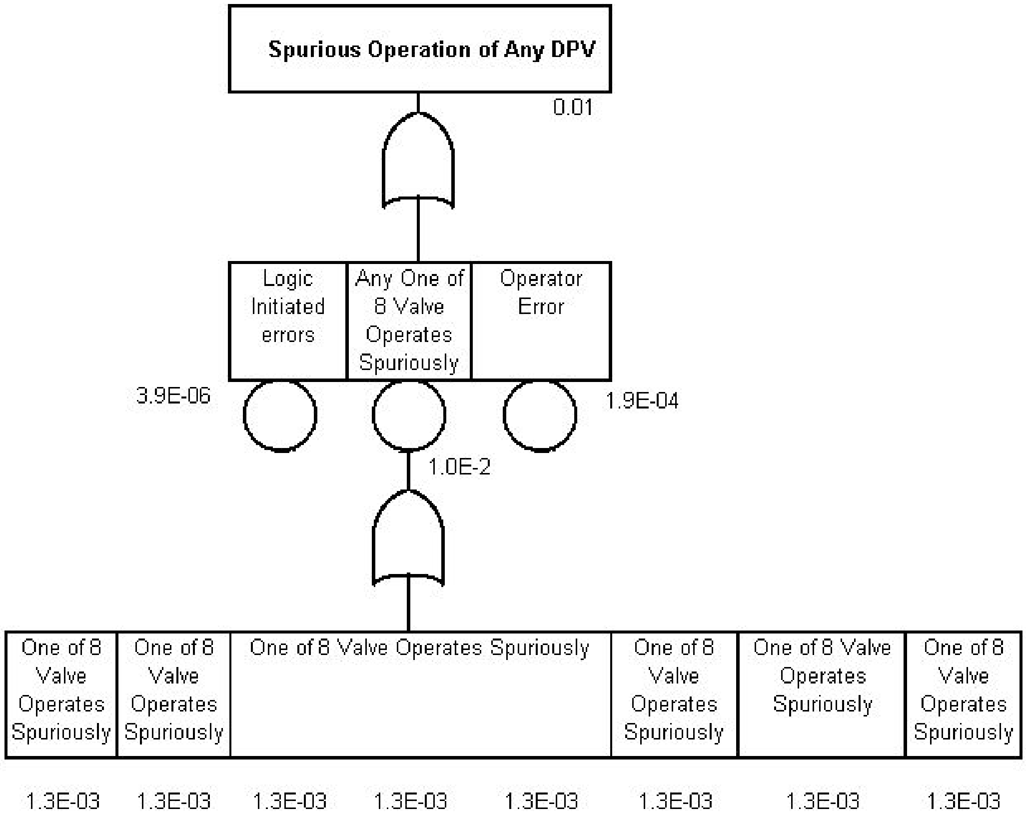


Figure 15A.6-12. Fault Tree for Inadvertent Opening of a DPV (IDPVO) – Overall Failure Probability

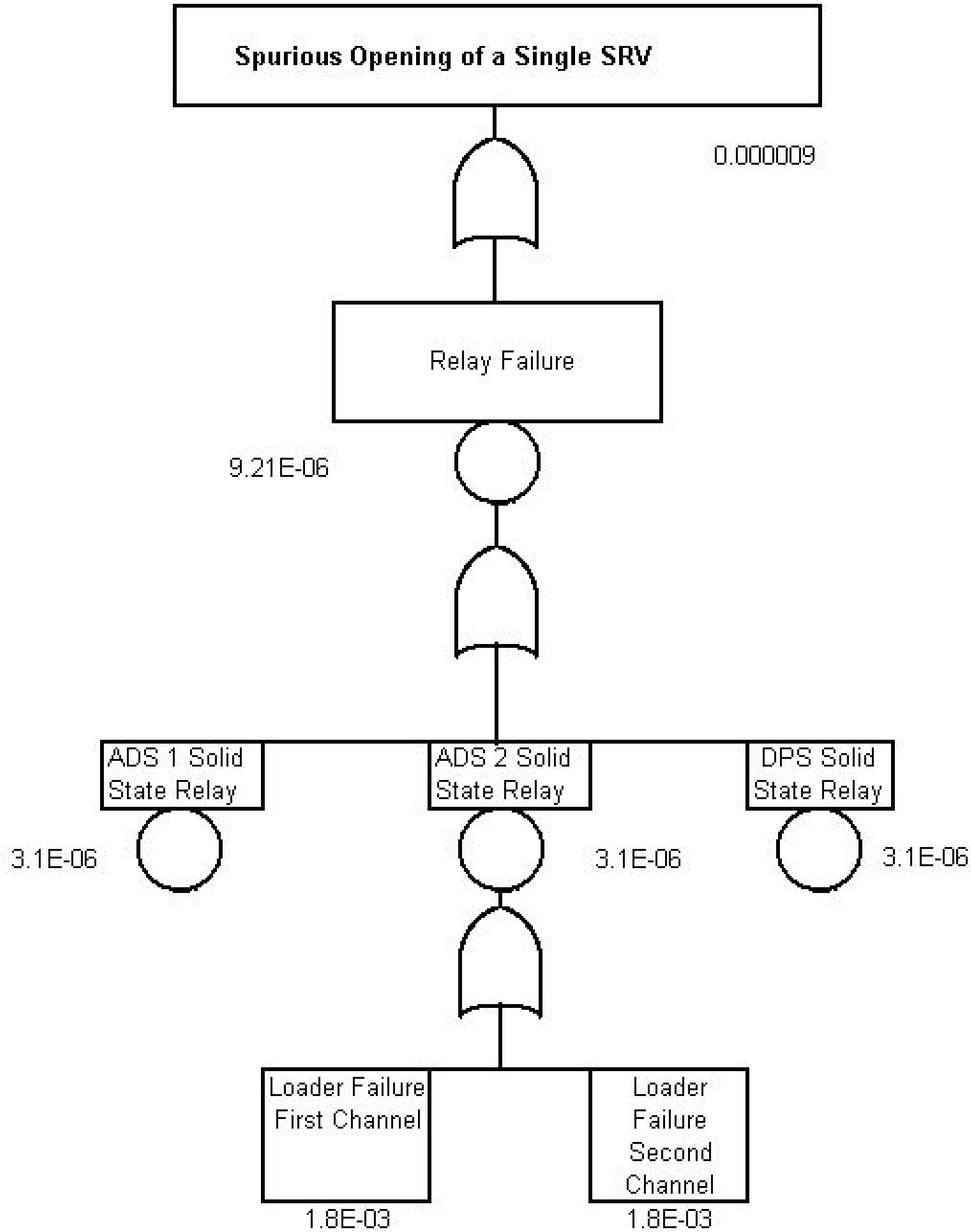


Figure 15A.6-13. Fault Tree for Dual Redundant Digital SOSRVO –Failure Probability For a Single Valve

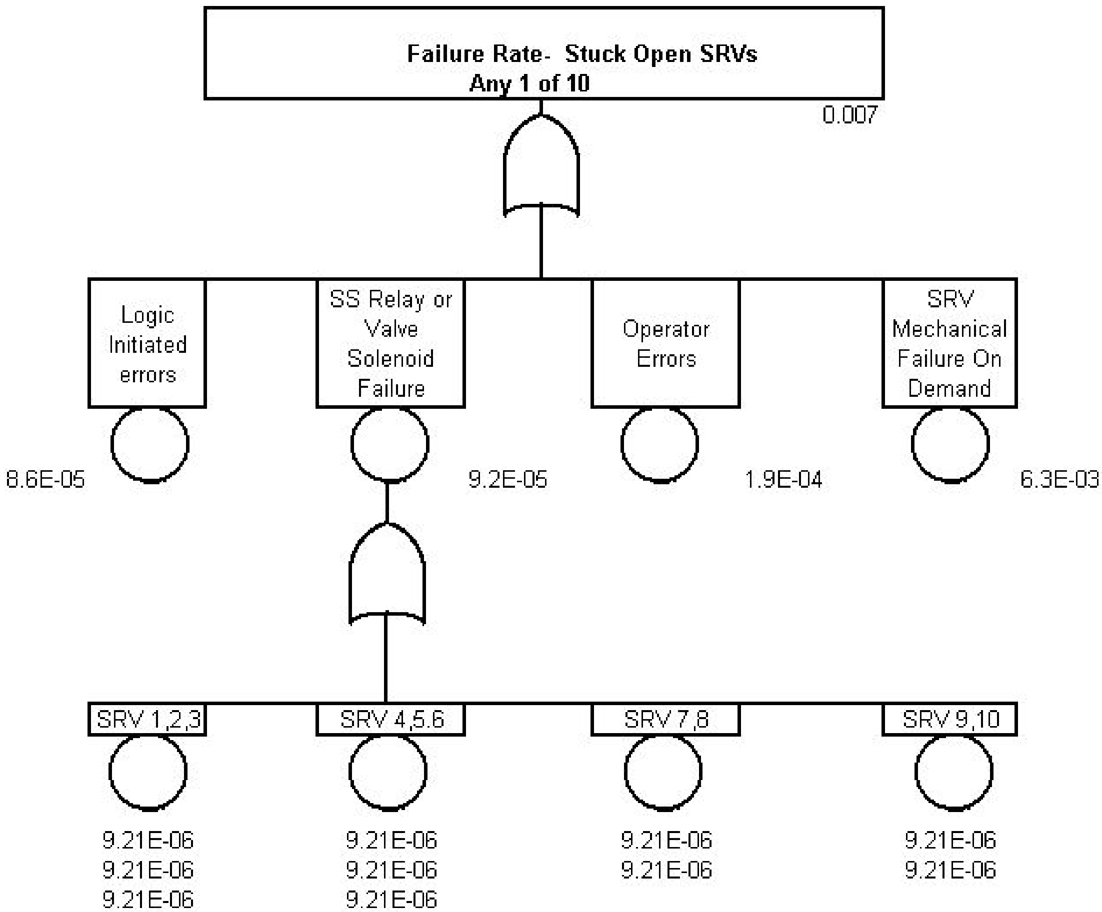


Figure 15A.6-14. Fault Tree for Stuck Open SRV SOSRV) – Overall Failure Probability

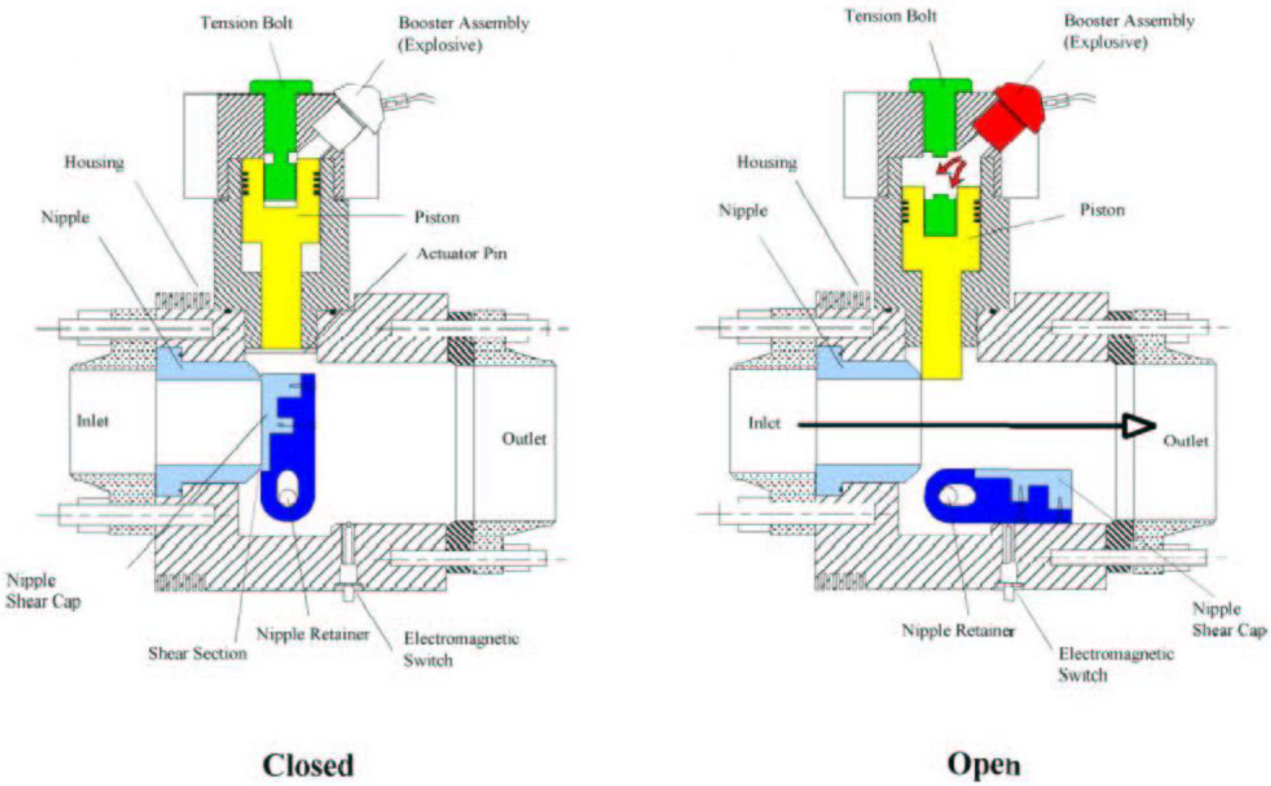
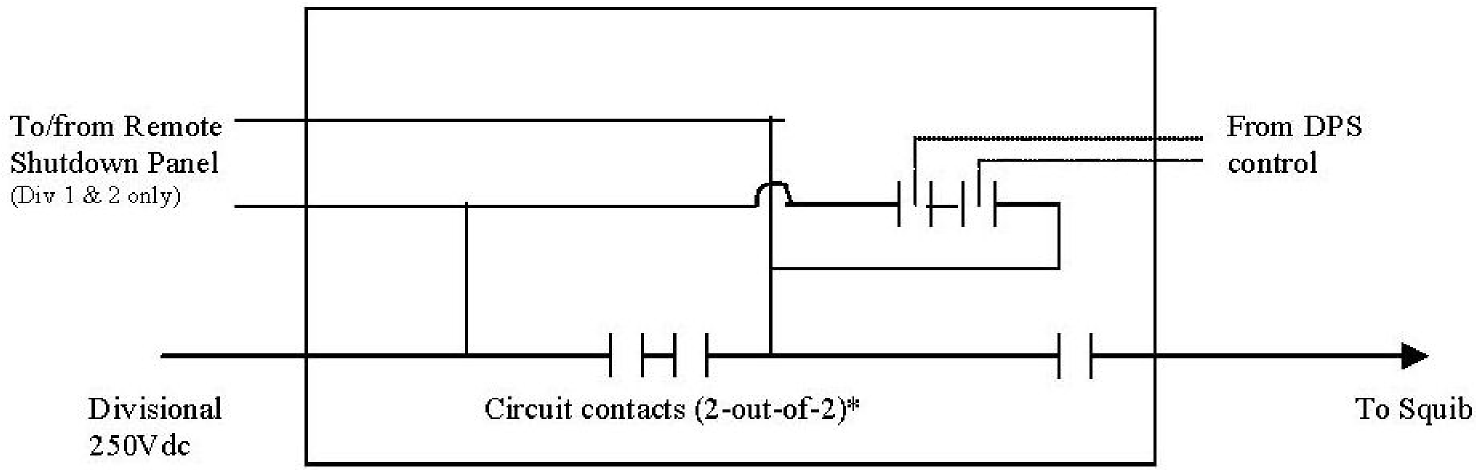


Figure 15A.6-15. Depressurization Valve



*Note:

- It needs 2 VLU outputs from the redundant dual VLU channels within the SSLC/ESF division to close the two contacts respectively.
- The bypass can be a keylock to disable the initiation of the squib by opening the circuit logic.
- DPS applies the trip in parallel with the ESF logic. It also needs two trip outputs to initiate the squib valve opening.

Figure 15A.6-16. Diverse Protection System Logic

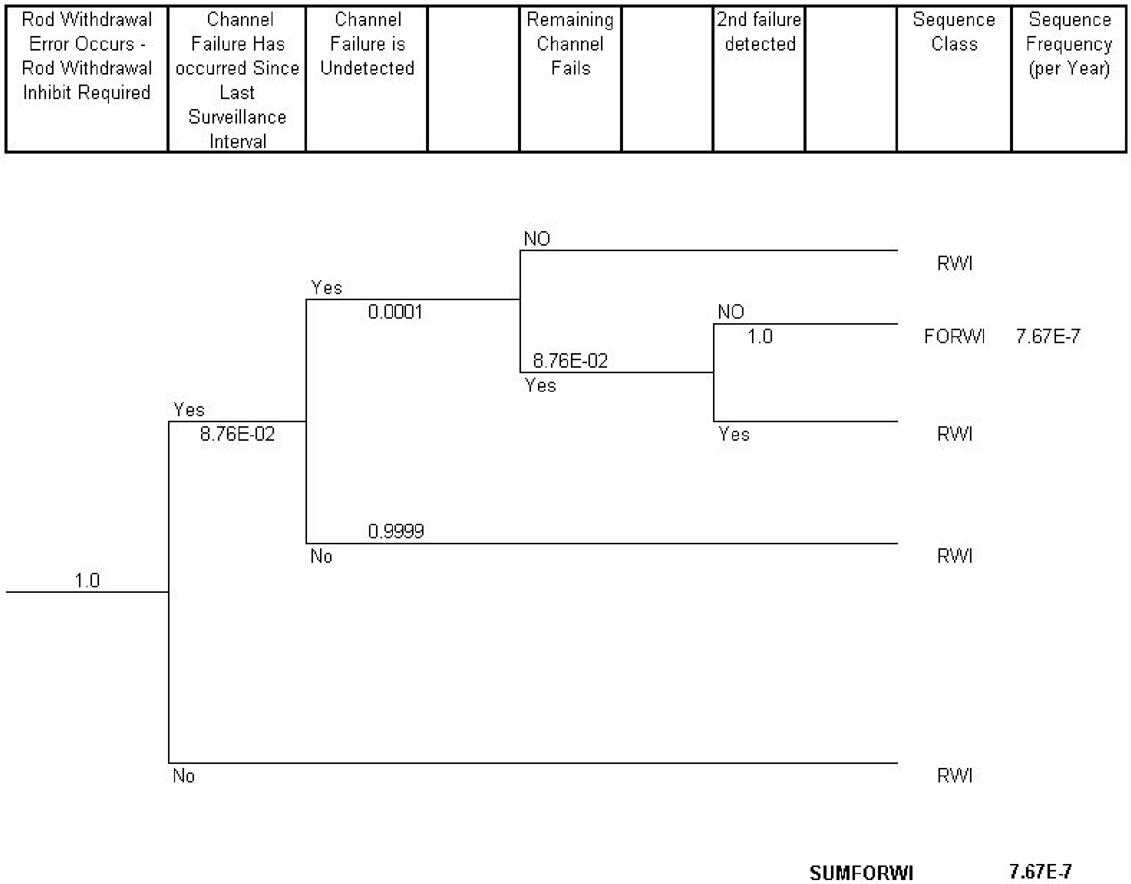


Figure 15A.7-1. Event Tree for Rod Withdrawal Error with A Failure of Rod Withdrawal Inhibit – Overall Failure Probability

15B. LOCA INVENTORY CURVES

This appendix provides additional detail on the distribution of iodine-131 for the design basis LOCA analysis found in Subsection 15.4.4. The information is in the form of a series of curves as is explained below.

Figure 15B-1 provides the total inventory of iodine-131 in various compartments as a function of time.

Figure 15B-1. Iodine Airborne Inventory in Primary Containment as a Function of Time